To: Leeds, NRR
Ref. G20110740

Jaegers, Cathy

From:	Lawrence Criscione [lscriscione@hotmail.com]
Sent:	Tuesday, November 08, 2011 6:29 PM
To:	Thadani, Mohan; Polickoski, James
Subject:	Please respond
Attachments:	10CFR2-206 Request on blocking LoTavg FWIS.pdf

cys: **EDO**
DEDMRT
DEDR
DEDCM
AO
RIV
OGC
Mensah, NRR
Banic, NRR
Scott, OGC
Bowman, OEDO

Mohan/Jim:

Please see the attached 10CFR2.206 petition which was copied to you on October 7, 2011.

It took me until October 14, 2011 to get a certificate so that I could electronically submit the attached request. Once I received this certificate an uploaded the petition, the NRC refused to accept it because some of the pages were scans. As a result, I re-uploaded the petition with the last 18 pages left out (i.e. I uploaded the first 40 of 58 pages).

The purpose of this email is to ensure that you are processing my FULL petition. The petition should be 58 pages long (a one page cover letter and a 57 page enclosure). If you have only processed the first 40 pages of my petition (i.e. the electronic submission from October 14th) then please add the last 18 pages of the attached document to my petition.

Please note, the attached document is what was copied to you on October 7, 2011. If you have already processed the Oct. 7 document, then there is nothing you need to do with this attachment.

Please respond to this email so that I know you have the entire petition. I am sorry for any confusion this has caused, but some of the documents included in the attachment were only available to me as scans.

Thanks,

Larry

Lawrence S. Criscione
(573) 230-3959

From: lscriscione@hotmail.com
To: bill.borchardt@nrc.gov
CC: [redacted]; jeanette.oxford@house.mo.gov; mohan.thadani@nrc.gov; james.polickoski@nrc.gov
Subject: FW: P-4/564°F FWIS at Wolf Creek and Callaway Plant
Date: Fri, 7 Oct 2011 23:00:31 -0400

Mr. Borchardt:

In late Spring 2010 I read Revision 01 to LER 482-2009-009. Because of work I spearheaded at Callaway Plant, upon reading the Wolf Creek Licensee Event Report I was concerned about the acceptability of the Reactor Shutdown procedure at Callaway Plant. In the summer of 2010 I wrote most of the attached 10CFR2.206 Request, which I did not submit since I believed that there was some likelihood that either Region IV or Callaway Plant would adequately respond to the Wolf Creek LER by revising Table 3.3.2-1 of the Callaway Plant Technical Specifications in a similar manner as Wolf Creek had applied to revise the same table in their Technical Specifications. However, it has now been 1½ years and I no longer think it is likely that Region IV or Callaway Plant are going to correct the issues with the Reactor Shutdown procedure and Callaway Plant will likely be again violating their Technical Specifications by bypassing the P-4/564°F FWIS in MODE 1 as part of their reactor shutdown plan for their upcoming refueling outage.

My request is:
The US NRC prevent Callaway Plant from bypassing the P-4/564°F FWIS in MODEs 1 through 3 until their Technical Specifications are revised to allow this practice.
I would like a preliminary evaluation of the steps in Callaway Plant's Reactor Shutdown procedure (which allow bypassing of the P-4/564°F FWIS) performed prior to Callaway Plant using that procedure to shut down the reactor for their upcoming refueling outage. The attached document provides the justification for this request as well as some less

Template: EDO-001 ML113l3A206 E-RIDS: EDO-01 eminent issues which need to be looked into.

A Senior Reactor Operator (SRO) at Callaway Plant [redacted] wrote a condition report (CARS 200703001) in March 2007 questioning whether or not the plant's Technical Specifications allowed the P-4/564°F FWIS to be bypassed in MODEs 1 & 2. Because this condition report was written just days prior to Refueling Outage 15, plant management pressured the Senior Reactor Operator to withdraw his condition report. Because the condition report was deleted prior to it being sent to the plant's Screening Committee, it does not appear in the Callaway Action Request System's database; however, I have a hard copy of the original report.

Enclosed with the attached letter is some background information regarding this issue. Included in the enclosure are some internal Callaway Plant emails concerning the plant's decision to allow bypassing the P-4/564°F FWIS in MODEs 1 & 2. From the emails it is apparent that there was much discussion regarding the decision of whether or not the plant's Technical Specifications allowed this safety function to be bypassed. The decision by Ameren to not pursue a change to Callaway Plant's Technical Specifications was a conscious decision by its Licensing Department. I am requesting that the US NRC review this decision and determine whether or not it is their opinion that a change to Callaway Plant's Technical Specifications is needed.

As a former submarine officer, I assume you are familiar with the following quote from Admiral Rickover:
A major flaw in our system of government, and even in industry, is the latitude allowed to do less than is necessary. Too often officials are willing to accept and adapt to situations they know to be wrong. The tendency is to downplay problems instead of actively trying to correct them.
I believe that if you choose to look into this issue, you will find that the tendency to "downplay problems instead of actively trying to correct them" was not only present within Ameren when they consciously chose not to revise their Technical Specifications prior to blocking P-4/564°F in MODE 1, but is also present in our own Region IV where they have allowed Callaway Plant to conduct practices, for which they cited Wolf Creek, because Ameren was able to get a less than adequate safety evaluation past NRR in the mid-1990s (a safety evaluation which only addressed bypassing the P-4/546°F FWIS in MODE 3 and was silent on MODEs 1 & 2).

I've copied Missouri legislator Jeanette Oxford on this email and the attached 10CFR2.206 Request. Representative Oxford has been assisting me with getting Safety Culture issues addressed at Callaway Plant, and she is also concerned with ensuring the ratepayers in the State of Missouri are not unnecessarily burdened with operating expenses stemming from poor stewardship of generating facilities (although the Wolf Creek Nuclear Operating Company is in Kansas, there may be some Missourians in the Kansas City area who fall into WCNOC's rate base since it is partially owned by Kansas City Power & Light). The Citizen's Utility Ratepayer Board in Kansas may be interested in the outcome of this request since this issue obviously concerns their ratepayers. It is my opinion that Callaway Plant has not been meeting Technical Specification 3.3.2; however, if I am wrong about Callaway Plant, then it is my opinion that Wolf Creek unnecessarily incurred expenses responding to the errors of NRC inspectors in 2009 and 2010. These expenses included protesting a noncited violation (NCV 05000482/2009004-04), writing and revising a Licensee Event Report (LER 482-2009-009, revisions 0 and 1), and processing a Technical Specification amendment (LA 194).

V/r,

Larry

Lawrence S. Criscione
(573) 230-3959
Human experience shows that people, not organizations or management systems, get things done.

From: Mohan.Thadani@nrc.gov
To: lscriscione@hotmail.com; James.Polickoski@nrc.gov
Date: Tue, 6 Sep 2011 08:19:07 -0400
Subject: RE: P-4/564°F FWIS at Wolf Creek and Callaway Plant

Larry:

I have not seen an amendment request, similar to the subject Wolf Creek Amendment, for Callaway Plant, Unit 1.

Mohan

From: Lawrence Criscione [mailto:lscriscione@hotmail.com]
Sent: Friday, September 02, 2011 8:27 PM
To: Thadani, Mohan; Polickoski, James
Subject: P-4/564°F FWIS at Wolf Creek and Callaway Plant

Jim/Mohan,

Please see the attached document (ML110550846) concerning the P-4/564°F FWIS at Wolf Creek.

Both Wolf Creek and Callaway Plant have a ESFAS feature wherein a Feed Water Isolation Signal is generated under the following conditions:

1. The reactor trip breakers are open (as read by permissive P-4) with P-4 not reset

AND

2. Reactor Coolant Temperature less than 564°F (Lo-Tavg).

On April 13, 2010 Wolf Creek Nuclear Operating Company a request (ML101100391) to amend its operating license such that the P-4/564°F FWIS was no longer required during MODE 3.

On March 30, 2011 we approved Wolf Creek's requested amendment (ML110550846).

To your knowledge, has Callaway Plant submitted a similar amendment? That is, to your knowledge, do the Technical Specifications at Callaway Plant allow it to block the P-4/564°F FWIS (function 8.a) during MODE 3?

Larry

Lawrence S. Criscione
(573) 230-3959

October 7, 2011

1412 Dial Court
Springfield, IL 62704

Bill Borchardt
Executive Director of Operations
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Mr. Borchardt:

I am submitting the information contained below as a 10CFR2.206 request. The address above is my home address; however, I work in the Washington, DC area and make it home to Illinois infrequently. Please send all correspondence to me electronically at either my personal email account (LSCriscione@hotmail.com) or my work email. If you must send me a hard copy, please send it to me at Mail Stop CSB/C2 A7.

The Reactor Shutdown procedure at Callaway Plant (OTG-ZZ-00005) is not compliant with the plant's Technical Specifications.

In 2007, the Reactor Shutdown procedure was revised to allow the operators to shut the plant down by tripping the control rods. Tripping the control rods causes the P-4 permissive to energize. One of the functions of the P-4 permissive is to enable a Feedwater Isolation Signal (FWIS) to occur on a "Low Tavg" signal (which occurs at 564°F). As part of the Reactor Shutdown procedure the Instrumentation & Controls technicians bypass the P-4/564°F FWIS.

Callaway Plant's Technical Specifications require the P-4 permissive and all its associated functions to be operable (i.e. not bypassed) when the plant's average coolant temperature is above 350°F. By bypassing both trains of the P-4/564°F FWIS, the plant is not in compliance with its Technical Specifications.

Please process this letter and its enclosure as a 10CFR2.206 request.

Very respectfully,

Lawrence S. Criscione

Lawrence S. Criscione, PE
(573) 230-3959

Enclosure (1)

Cc: Jeanette Mott Oxford, Missouri House of Representatives

10CFR2.206 Request Regarding Blocking of the P-4/LoTavg Feedwater Isolation Signal (FWIS) at Callaway Plant

§1. Background

Wolf Creek is a nuclear reactor plant near Burlington, Kansas which is operated by the Wolf Creek Nuclear Operating Company (WCNOC). Callaway Plant is a nuclear reactor plant in Callaway County, Missouri which is operated by Ameren Corporation. The two nuclear reactor plants are Westinghouse 4-Loop Pressurized Water Reactors and are of similar vintage and design.

As part of their Engineered Safety Features Actuation System (ESFAS) both plants have a permissive denoted P-4. P-4 is an electrical signal which is energized when the reactor trip breakers are open and which permits the completion of other electrical signals. One of the several electrical signals which are permitted by P-4 becoming energized is a Feedwater Isolation Signal (FWIS) on low average reactor coolant temperature (LoTavg). In 2007 (and possibly to this day) LoTavg at Callaway Plant was electrically set to 564°F.

The Reactor Trip Breakers (RTBs) provide power to the reactor's control rods. The opening of the RTBs cause the mechanical equipment holding the control rods out of the reactor to de-energize and thereby cause the control rods to fall into the reactor core. The control rods are made of a neutron absorbing material and their insertion into the reactor core disrupts the nuclear fission chain reaction, causing the reactor to shut down.

Pressurized Water Reactors (PWRs) in the United States are inherently designed such that they will automatically shut down due to a sharply rising reactor coolant temperature. The corollary to this design feature, however, is that a sharply lowering reactor coolant temperature can, in conjunction with certain equipment failures, cause a shutdown reactor to inadvertently restart. One of the "defense-in-depth" measures designed into Wolf Creek and Callaway Plant to prevent a sharply lowering reactor coolant temperature on a shutdown reactor is the Feedwater Isolation Signal which occurs when average reactor coolant temperature drops below 564°F with the reactor trip breakers open (i.e. the P-4/564°F FWIS).

A Feedwater Isolation Signal causes the plant's normal feedwater path to isolate and the plant's auxiliary feedwater system to activate. Although this is a desirable outcome during many reactor accident scenarios, operating on auxiliary feedwater does have its drawbacks:

- Controlling steam generator levels is more difficult for the operators while operating on auxiliary feedwater than while operating on normal feedwater

* Auxiliary feedwater is typically cooler than normal feedwater and can cause thermal stresses to the feedwater piping.

For the above two reasons, by the mid-1990s both Callaway Plant and Wolf Creek were in the habit of using electrical jumpers to bypass the P-4/564°F FWIS during certain plant evolutions. Note that it is not my opinion that this was a bad practice. It is my opinion that this was a good practice, however, the acceptability and control of this activity may not have been adequately evaluated by the utilities and the US Nuclear Regulatory Commission to determine if the benefits of the practice (e.g. less thermal stress and better operator control) outweighed the drawbacks (e.g. degraded "defense-in-depth" for certain unlikely combinations of equipment failures).

§1.1. Callaway Plant Operating License Amendment 126

In 1996, Callaway Plant originated an internal modification package (CMP 96-1016A) to install bypass switches around the P-4/564°F FWIS so that, when plant evolutions desired this signal to be bypass, the signal could be bypassed by use of installed switches instead of by installing jumpers. This was a wise modification. Installing jumpers on engineered safety features always involves some amount of risk of human error that is not present during the operation of an installed switch.

As part of the implementation of CMP 96-1016A, Union Electric (Ameren's predecessor) applied to the US NRC for a change to their Technical Specifications on August 8, 1997. This application was later supplemented on November 10, 1997 and approved as License Amendment number 126 (LA126) on April 23, 1998. The approval letter is located in the Agencywide Documents Access and Management System (ADAMS) as ML021640348. Enclosure 2 of this letter (the safety evaluation for LA 126 conducted by the Office of Nuclear Reactor Regulation) is included below on pp. 40-42.

During 1996-98 [redacted], a highly-experienced and thorough US NRC licensed Senior Reactor Operator (SRO) at Callaway Plant, was involved in the review and implementation of LA126 and CMP 96-1016A.

§1.2. 10CFR50.59 Screening of the Callaway Plant Reactor Shutdown Procedure

I was a US NRC licensed Senior Reactor Operator at Callaway Plant when, in January 2005, the Outage Manager [redacted] approached me about revising the Reactor Shutdown Procedure (OTG-ZZ-00005) such that it permitted manually tripping the reactor as one of the normal means of conducting a reactor shutdown. Due to my involvement in revising the Plant Cool Down procedure (OTG-ZZ-00006) and assignments for the Steam Generator Replacement Outage (RF14) it took me nearly a year before I could act upon [redacted] request.

In early January 2006 I originated Request For Resolution (RFR) 200600140 in the Callaway Plant Action Request System (i.e. Ameren's computerized system for implementing the legally mandated Problem Identification and Resolution process at Callaway Plant). As originally written, RFR 200600140 requested resolution of two questions:

- *Do current licensing documents require the P-4/Lo Tavg FWIS to be operable in MODE 1 or 2 (the originator of this RFR could not find any requirement)?*
- *Do any Operability Determinations rely on the P-4/Lo Tavg FWIS being available?*

On January 10, 2006 one of the engineering supervisors [redacted] requested that I re-write RFR 200600140 such that it was more general (see below, p. 39 of this enclosure). I no longer have access to RFR 200600140, but the wording was something to the effect of "make any licensing changes necessary for revision of the Reactor Shutdown procedure". RFR 200600140 was then rejected by the RFR Screening Committee as being too general.

In April 2006, the issue was re-submitted as RFR 200602749. Although this document was only concerned with blocking the FWIS permitted by P-4 and NOT with disabling permissive P-4, RFR 200602749 was closed with a Lead Response which merely discussed the fact that P-4 is required to be operable in MODEs 1 and 2 per Table 3.3.2-1 of the plant's Technical Specifications (see below, p. 30 of this enclosure). Note that at the time (and quite possible still) Callaway Plant did not require any closure review of Requests for Resolution to ensure they were adequately addressed.

On September 5, 2006 I wrote yet a third Request for Resolution on this topic (RFR 200607357) which made the follow requests (see below, p. 29 of this enclosure):

- either *document that there are no regulatory requirements which prevent bypassing the Feed Water Isolation Signal caused by P-4 and Lo Tavg (546°F) prior to tripping the reactor from MODE 1 or 2,*
- or *make any necessary amendments to license documents to allow bypassing the P-4/Lo Tavg FWIS prior to tripping the reactor from MODE 1 or 2.*

In a September 7, 2006 email exchange regarding "*Tripping the Control Banks in OTG-ZZ-00005*" (see below, pp. 26-28 of this enclosure), [redacted] of Callaway Plant's Licensing group wrote, in reference to bypassing the P-4/564°F FWIS in MODEs 1 and 2, that:

> [redacted] *and his group* [Safety Analysis] *don't support doing just a Bases change.*

In a later email that same morning, [redacted] stated:

I maintain this [bypassing the P-4/564°F FWIS in MODE 1 or 2] *can be done via a Bases change only; P-4 is operable whenever it receives the required RTB position inputs and the permissive is satisfying its logic outputs to SSPS. The enabled function that relies on P-4, FWIS on low T-avg, is not required by ESFAS Function 5 nor credited in any accident analysis that I'm aware of.*

[redacted] then responded in an email stating (see below, p. 26 of this enclosure):

The Safety Analysis group is not opposed to a TS Bases change. Our point was that we interpreted the TS such that a TS change would be required. However, we deferred to Licensing as the TS Subject Matter Experts. If [redacted] [Licensing] *believes and can document why a TS change is not needed and that only a TS Bases change is sufficient, we have no strong objections.*

Later, in a September 22, 2006 email (see below, p. 25 of this enclosure), [redacted] makes statements which seem to endorse [redacted] view that a TS change was required:

...I'd recommend a footnote be added to Table 3.3.2-1 Function 8a describing the circumstances behind our desire to block this particular enabled function from P-4 (i.e., allow it to be blocked during any of the Applicable MODES for P-4 during a plant shutdown only, to be restored prior to MODE 2 entry ascending). [Note that this is what Wolf Creek was required to do in 2010 in order to continue to block the P-4/564°F FWIS during MODE 3.]

Safety Analysis makes a valid point when they say typical ITS rules of usage do not allow the Bases to modify the LCO Applicability.

...I think the best way to resolve all of the above is to submit a TS change and get a very clear thumbs up or down from NRC.

[redacted] position, however, changed by January 9, 2007 when he stated in an email (see below, p. 22 of this enclosure):

In the NRC's Safety Evaluation for LA 126 dated 4-23-98, they specifically reviewed and found acceptable our bypass switch design and our using it to block the FWIS initiated by the coincidence of P-4/low T-avg as long as its use was limited to the following plant conditions:

- *with T-avg less than or equal to 564°F (you can be in MODE 1 or 2, but must be ≤ 564°F)*
- *just prior to opening the RTBs (which satisfies the P-4 portion of this feedwater isolation signal's logic).*

NRC also wanted this FWIS restored by defeating the bypass prior to entering MODE 2 ascending during startup from an outage. As long as these limitations are observed, no amendment is needed since it's already been reviewed and approved by NRC. We could have been doing this since 4-23-98.

By January 10, 2007, I had been trying unsuccessfully for over a year to get the Callaway Plant organization either to state in some type of Quality Assurance record that it was alright for Operations to bypass the P-4/564°F FWIS when manually tripping the reactor in MODE 1 or 2, or to apply to the US NRC to change Table 3.3.2-1 of the plant's Technical Specifications in order to permit the bypassing of the P-4/564°F FWIS in MODEs 1 and 2 just prior to manually tripping the reactor. It was at this point that the issue had, after nearly two years, come full circle back to [redacted] (the [redacted] who in 2005 originally requested that the Reactor Shutdown procedure be modified to allow manually tripping the reactor) who, by 2007, was the [redacted]. In a January 10, 2007 email to [redacted] (see below, pp. 21-22 of this enclosure) I stated:

I would like to see it documented on a QA record that Accident Analysis and Licensing concur that the P-4/564°F FWIS can be bypassed in MODE 1 below 564°F.

In an effort to meet the above request, it was decided in a January 10-11, 2007 email exchange (see below, pp. 18-20 of this enclosure) that:

...the 10CFR50.59 screening evaluation for the procedure change is the appropriate place to document that this change is within Callaway's current licensing bases.

On February 28, 2007 [redacted] of Licensing prepared the 10CFR50.59 Screening for Revision 00 to OTG-ZZ-00005, Addendum 01 (the new procedure which allowed bypassing the P-4/564°F FWIS just prior to manually tripping the control rods). [redacted] of Safety Analysis reviewed this 10CFR50.59 Screening on March 1, 2007. In the screening document it is stated (see below, p. 45 of this enclosure):

This non-critical enabled function, FWIS on P-4 coincident with low RCS T-avg, is not a TS required SSC. If it were a TS required SSC, it would be required to be listed as a sub-function under TS Table 3.3.2-1 Function 5. It is not. FWIS on P-4 coincident with *low RCS T-avg does not meet any of the four criteria for TS inclusion in 10CFR50.36 (c)(2)(ii).*

At the time (i.e. early March 2007), I failed to question the above assertions since they supported my goal of implementing the new Reactor Shutdown procedure prior to the upcoming refueling outage (RF15 in April 2007).

§1.3. Callaway Action Request 200703001

On March 29, 2007 I gave [redacted] a copy of OTG-ZZ-00005, Addendum 01 to review. That night, [redacted] was the Field Supervisor. I can no longer recall why I gave [redacted] a copy of the procedure. It may have been that his crew was scheduled to shut down the reactor on April 1, 2007 to start Refueling Outage 15. Or it may have been that, as an experienced Senior Reactor Operator, I wanted his opinion on the new procedure.

[redacted] had a significant concern with OTG-ZZ-00005, Addendum 01. In his opinion, this new procedure violated Technical Specification 3.3.2. Unbeknownst to me until that evening, [redacted] had been involved with modification package CMP 96-1016A and Operating License Amendment 126. From [redacted] involvement in these activities it was his opinion that when ULNRC-03681 was written the phrase "in a shutdown evolution" meant the rod banks were inserted and the operators were at the point of opening the reactor trip breakers (i.e. the operators were performing the evolution of shutting down the plant and were already in MODE 3). It should be noted that when LA 126 was issued, the only way to shut down the reactor at Callaway Plant per the normal reactor shutdown procedure was to manually drive the control rods into the reactor core. Therefore, when the P-4/564°F FWIS was to be bypassed prior to opening the reactor trip breakers, the control rods were already fully inserted and the reactor unquestionably shut down and in MODE 3. However, with the release of Addendum 01 of OTG-ZZ-00005 in March 2007 it was now going to be the practice to bypass the P-4/564°F FWIS with the reactor in MODE 1, with the nuclear fission reaction still critical, and with the control rods either fully or near fully withdrawn. It was [redacted] view that License Amendment 126 did not apply to bypassing the P-4/564°F FWIS under the new conditions.

I was in no position to either refute [redacted] concerns or to act upon them and change the plan for the upcoming reactor shutdown (which was to occur on the evening of April 1, 2007). My advice to [redacted] was to document his concerns in the Callaway Action Request System, which he did as CAR 200703001. To ensure that CAR 200703001 got the level of concern which it warranted, I informed all interested parties about it in an email in the early morning hours of March 30, 2007 (see below, p. 16 of this enclosure).

[redacted] concerns were not well received (see below, pp. 13-15 of this enclosure). For my part, after spending the better part of three years optimizing the Reactor Shutdown and Plant Cool Down procedures, I was accused by Operations of attempting to sabotage the refueling outage. By the nine o'clock hour on March 30, 2007 I was tired of arguing with my superiors, and I was willing to accept and adapt to a situation I knew to be wrong and to focus on downplaying problems instead of actively trying to correct them. In an email to [redacted] (see below, p. 13 of this enclosure) I offered to answer CAR 200703001 with wording similar to the 10CFR50.59 Screening of Revision 00 to OTG-ZZ-00005, Addendum 01. Similarly broken of his desire to continue fighting, [redacted] deleted CAR 200703001 (see below, p. 13 of this enclosure). Since CAR 200703001 had never been sent to the daily Screening Committee meeting, there is no record of it at Callaway Plant, but somewhere in a box in my vacant home in Jefferson City, Missouri I still have a hardcopy of it.

It should be noted that beating down [redacted] and his inconvenient safety concern was financially the correct course of action for Ameren to take under the US NRC's Reactor Oversight Process (ROP). Under the Reactor Oversight Process, the safety significance of an issue determines the level of regulatory scrutiny and punishment which will be applied to it. As [redacted] would assuredly acknowledge, his concerns were of low safety significance due to the fact that the P-4/564°F FWIS would be bypassed for a very short time and the likelihood of the requisite equipment failures occurring during that short time window is very slight. [redacted] concerns were about doing the right thing from a regulatory standpoint. However, in order to do the right thing, Callaway Plant would have needed to change its refueling outage schedule at the eleventh hour. At hundreds of thousands of dollars an hour, even slight changes to the outage schedule would have amounted to significant expenses. Bonuses at Callaway Plant are heavily dependent on meeting refueling outage schedules and costs, so there was monetarily much at risk by Callaway Plant's upper management. Conversely, under the Reactor Oversight Process, low risk-significant violations of plant licensing commitments typically amount to no monetary penalties. Although "willful violations" of licensing commitments are treated seriously, these are nearly impossible to objectively prove. The odds that this issue would ever be brought before the US NRC were slight (it is only by chance that I came across it again when reviewing revision 01 to Wolf Creek's LER 482-2009-009). Now that it has been brought before the NRC, it is unlikely this 10CFR2.206 will be acted upon. Even if acted upon, it is still very unlikely that there will be anything greater than a noncited violation issued to Callaway Plant. By removing the subjective judgments of the regional leadership from the regulatory formula, the Reactor Oversight Process has made it nearly impossible to punish plants who are "gaming the regulations" until a risk significant incident occurs; as a result, we are reactive in our regulation instead of proactive.

§1.4. Integrated Inspection Report at Wolf Creek Nuclear Operating Company

On August 22, 2009 US NRC inspectors at Wolf Creek observed Instrumentation and Controls (I&C) technicians install jumper wires to bypass the P-4/564°F FWIS while the plant was in MODE 3. In Integrated Inspection Report 05000482/2009004 (see below, pp. 47-51 of this enclosure) the inspectors stated:

> *The inspectors and the NRR technical specification branch found this to be contrary to the Updated Safety Analysis Report, the technical specifications, the technical specification bases, and the NRC safety evaluations supporting the technical specifications.*

Based on the above observation, the inspectors issued a noncited violation (NCV 05000482/2009004-04) for "Failure to Implement Engineered Safety Features Actuation System Technical Specifications Results in a Missed Mode Change."

§1.5. Licensee Event Report 482-2009-009

As a result of NCV 05000482/2009004-04, Wolf Creek Nuclear Operating Company submitted Licensee Event Report 482-2009-009 in December 2009 and submitted a revision (LER 482-2009-009-01) on March 22, 2010.

Item I.C.5 of the TMI Action Plan requires that licensees shall:

> *...prepare procedures to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs.*

It is unclear how Callaway Plant meets the above requirement. One would think that a Licensee Event Report from Callaway Plant's "sister plant" concerning an entry into Technical Specification 3.0.3 would be "*operating information pertinent to plant safety*". It is unclear why, following the release of LER 482-2009-009, no one at Callaway Plant questioned the plant's practice of bypassing the P-4/564°F FWIS not only in MODE 3 but also in MODEs 1 and 2.

§1.6. Wolf Creek Nuclear Operating Company Operating License Amendment

On April 13, 2010 Wolf Creek Nuclear Operating Company applied to the US NRC for an amendment to their Technical Specifications to allow bypassing the P-4/564°F FWIS during MODE 3 (see below, pp. 54-56 of this enclosure). On March 30, 2011 the US NRC approved amendment 194 to Wolf Creek's Technical Specifications. Amendment 194 added a footnote to TS Table 3.3.2-1 similar to what [redacted] proposed for Callaway Plant in an email on September 22, 2006 (see below, p. 25 of this enclosure).

§1.7. Willful Violations

From January 2006 through February 2007 there was a fair amount of discussion and debate at Callaway Plant regarding whether or not an amendment to the plant's Technical Specifications was required to allow bypassing the P-4/564°F FWIS in MODEs 1 and 2 just prior to manually tripping the reactor. In the end, it is my opinion that [redacted] (of Licensing) and [redacted] (of Safety Analysis) reached the wrong conclusion in their 10CFR50.59 screening of Revision 00 to OTG-ZZ-00005, Addendum 01. However, it is also my opinion that neither one of these men were guilty of either willful violation or incompetence; their error was nothing more than an honest mistake, made while trying to understand complex licensing requirements (e.g. although TS Table 3.3.2-1 clearly requires P-4 to be operable, since [redacted]

Although I do not believe there was willful violation committed during the 10CFR50.59 review process, I do believe that either willful violations or instances of gross incompetence have occurred at Callaway Plant since February 2007 with regard to bypassing the P-4/564°F FWIS. Specifically:

- [redacted]
- [redacted]

It is my opinion that the two bulleted items above demonstrate that Callaway Plant has either a grossly incompetent management team or has a culture which willfully chooses to ignore inconvenient licensing issues.

There are some at the NRC who will attempt to prevent the above items from being transparently addressed through the 10CFR2.206 process. If the concerns above are remitted to another process (e.g. the anti-transparent Allegation Process) I believe that it is inappropriate for inspectors from Region IV to be assigned to investigate these concerns. Region IV has consistently validated Callaway Plant's Problem Identification and Resolution (PI&R) process and Safety Culture as satisfactory and therefore has a vested interest in downplaying the problems which exist in Ameren's corporate culture. It is my opinion that any allegation investigation should be performed by NRC Headquarters or another regional office.

§2. Requests per 10CFR2.206

Please treat the requests in the sections below per 10CFR2.206.

§2.1. Immediate Action Request

I request that the US Nuclear Regulatory Commission take the following action in a timely manner to ensure that Callaway Plant does not inadvertently enter Technical Specification 3.0.3 by violating Technical Specification 3.3.2:

1. Prohibit Callaway Plant from bypassing the P-4/564°F Feedwater Isolation Signal until the practice has been reviewed by the US NRC and determined to be in compliance with Technical Specification 3.3.2.

§2.2. Requests for Evaluation

I request the appropriate staff at the US NRC Office of Nuclear Reactor Regulation perform the following actions in order to evaluate my concerns:

2. The US NRC review the Green Noncited Violation of Technical Specification 3.0.3 from August 22, 2009 contained on pages 3, 4, 19, 20 and 21 of the enclosure to Integrated Inspection Report 05000482/2009004 (ML093140803) and determine if a similar violation applies to Callaway Plant (these pages are provided below as pp. 47-51).
3. The US NRC review LER 482-2009-009-01 (ML100890421) and determine if a similar LER is required by Callaway Plant to report any violations of TS 3.0.3 as a result of their bypassing of the P-4/564°F FWIS during MODEs 1 or 2.
4. The US NRC review the Green Noncited Violation of Technical Specification 3.0.3 from August 22, 2009 contained on page 10 of the enclosure to Integrated Inspection Report 05000482/2009005 (ML100430713) and determine if a similar violation applies to Callaway Plant (this page is provided below as p. 52).

5. The US NRC review Amendment 126 to Callaway Plant's operating license (ML021640348) and determine if they believe there is anything in this license amendment which allows the utility to block the P-4/564°F Feedwater Isolation Signal during MODEs 1 or 2 just prior to shutting down the reactor by manually tripping the control rods. Please comment specifically on paragraph 2.4 of Enclosure 2 (found on page 24 of ML021640348 and provided below on p. 41).
6. The US NRC review Callaway Plant's Reactor Shutdown Procedure (OTG-ZZ-00005) including the 10CFR50.59 screening paperwork for OTG-ZZ-00005, Addendum 01, Revision 00 (included below as pp. 43-46) which was signed [redacted] (2/28/2007) and [redacted] (3/1/2007), and determine if the US NRC agrees with the utility's answer to screening question 5. Please comment specifically on the statement:

    This non-critical enabled function, FWIS on P-4 coincident with low RCS T-avg. is not a TS required SSC. If it were a TS required SSC, it would be required to be listed as a sub-function under TS Table 3.3.2-1 Function 5. It is not. FWIS on P-4 coincident with low RCS T-avg does not meet any of the four criteria for TS inclusion in 10CFR50.36(c)(2)(ii).

§2.3. Requests for Action

Based on the determinations made by the NRC staff for items 1-6 in section §2.2, please take action per either section §2.3.1 or §2.3.2 as appropriate.

§2.3.1. Actions regarding Callaway Plant

If the NRC determines that by-passing the P-4/564°F Feedwater Isolation Signal during MODEs 1 and 2 at Callaway Plant is a violation of the plant's Technical Specifications, then I request that the actions below be taken. Note that, due to Region IV's past involvement with this issue, I suggest the actions below be handled by inspectors and/or investigators from either headquarters or a different regional office:

7. Issue a violation to Callaway Plant for every inadvertent entry into TS 3.0.3 which has occurred as a result of by-passing the P-4/564°F FWIS during MODEs 1 and 2.
8. Determine what deficiencies in Callaway Plant's 10CFR50.59 Screening Process allowed a procedure change to be made which violated the plant's Technical Specifications.
9. Review the email trail included in this enclosure (pp. 13-39) and investigate what failed in the Safety Culture at Callaway Plant that caused the concerns raised in Callaway Action Request 200703001 to go unaddressed.
10. Determine if there are any deficiencies in Callaway Plant's ability to process and learn from industry Operating Experience (OpE) in light of the fact that apparently no action was taken by Ameren in response to LER 482-2009-009 revisions 00 and 01.

11. Determine if there are any deficiencies in Callaway Plant's ability to work with industry peers in light of the fact that their "sister plant" submitted a License Amendment (Wolf Creek's LA 194) which, although somewhat applicable to Callaway Plant, was not addressed by Callaway Plant.
12. Determine why the US NRC did not look at Callaway Plant's practices regarding blocking the P-4/564°F FWIS once it was noted in August 2009 that Wolf Creek's practices (Callaway's "sister plant") did not meet her Technical Specifications.

§2.3.2. Actions regarding Wolf Creek Nuclear Operating Company

If the NRC determines that by-passing the P-4/564°F Feedwater Isolation Signal during MODEs 1 and 2 at Callaway Plant is not a violation of the plant's Technical Specifications, then I request that the actions below be taken. Note that, due to Region IV's past involvement with this issue, I suggest the actions below be handled by inspectors and/or investigators from either headquarters or a different regional office:

13. Review NCV 05000482/2009004-04 in light of the Callaway Plant determination and, if appropriate, withdraw this noncited violation (see below, p. 51 of this enclosure).
14. Review the NCV from IR 05000482/2009005 regarding LER 482-2009-009-00 in light of the Callaway Plant determination and, if appropriate, withdraw this noncited violation (see below, p. 52 of this enclosure).
15. Review Licensee Event Reports 482-2009-009-00 and 482-2009-009-01 in light of the Callaway Plant determination and, if appropriate, have Wolf Creek either withdraw the LERs or submit a new revision which correctly discusses how her Technical Specifications were not met.
16. If appropriate, reimburse Wolf Creek Nuclear Operating Company for any expenses unnecessarily incurred in submitting and processing LER 482-2009-009 revisions 00 & 01 and Amendment 194 to the plant's Technical Specifications so that the nuclear rate payers of the State of Kansas are not unfairly burdened by errors made by the staff of the US Nuclear Regulatory Commission.

§3. Supporting Documents

The remainder of this enclosure is supporting documentation:

From: Criscione, Larry S.
Sent: Sunday, April [redacted]
To: [redacted]
Subject: FW: CAR [redacted]

Issue resolved.

From: Criscione, Larry S.
Sent: Friday, March 30, 2007 9:58 PM
To: [redacted]
Subject: RE: CAR 200703001

Dave,

[redacted] is satisfied we can defend all our NRC commitments associated with the bypass switches of the P-4/564°F FWIS and has deleted CARS 200703001 based on the following:

1. Licensing has documented in the 10CFR50.59 Screening of OTG-ZZ-00005, Addendum 01 that the P-4/564°F FWIS may be used in MODE 1 or 2
2. [redacted] email below indicates Licensing's basis for the 10CFR50.59 Screening

If you would still like this addressed in the CAR system I can add it to the Refuel 15 critique. It was obvious today that [redacted] did not appreciate being questioned at the "eleventh hour". I apologize for this. I did my best to ensure issues such as [redacted]s were identified and resolved earlier in the cycle but unfortunately did not succeed.

V/r,
Larry

From: Criscione, Larry S.
Sent: Friday, March 30, 2007 9:03 PM
To: [redacted]
Subject: RE: CAR [redacted]

[redacted]

I found OL 126. See attachment.

Section 2.4 used the same wording as ULNRC-03681: "during startup and shutdown evolutions with Tavg ≤ 564°F just prior to opening the reactor trip breakers."

Although this wording does not specifically state the function can be used in MODE 1 or 2, the wording neither specifically states the function can only be used in MODE 3 or below.

I am growing tired of this issue too, but don't let that discourage you if you still have a concern. With our current level of regulatory scrutiny on design basis, we need to make sure we have all our commitments addressed.

My opinion is the wording of "during startup and shutdown evolutions with Tavg ≤ 564°F just prior to opening the reactor trip breakers" can be interpreted to apply in MODE 1 or 2 as well as in MODE 3. From [redacted] email below, Licensing is obviously comfortable defending this.

Please look at OL 126 (it's short) and let me know if you have any further concerns. If you want this issue documented in a CARS response vice an email, I can answer the CARS.

Thanks,
Larry

From: [redacted]
Sent: Friday, March 30, 2007 7:51 PM
To: Criscione, Larry S.; [redacted]
Cc: [redacted]
Subject: RE: CAR 200703001

LA126 only requires that T-avg be less than or equal to 564: use of the block switch isn't tied to any MODE. Restoration is tied to MODE 2 entry. Neither our ULNRC submittals nor LA126 were contingent on the insertion of control/shutdown banks, or lack thereof. Rod position was never discussed. NRC wanted the use of this block switch to be limited to startups and shutdowns at or near the time the FWIS coincidence is made up for this function.

If you don't believe me, rescind OTG-ZZ-00005 Addendum 1. I grow tired of these never-ending email daisy chains.

[redacted]

From: Criscione, Larry S.
Sent: Friday, March 30, 2007 7:41 PM
To: [redacted]
Cc: [redacted]
Subject: RE: CAR 200703001

[redacted]

I re-reviewed the 50.59 Screening last night when [redacted] raised his concerns.

I have never had any concern with the T/S bases for P-4. I have always agreed with the 50.59 Screening with regard to characterization of P-4/564°F. Additionally, I have never had any concern with the safety of performing OTG-ZZ-00005, Addendum 01as it is written (the title of my email was over-stated). To me the issues brought forward by [redacted] are entirely a matter of second checking that all our regulatory commitments are met.

[redacted] was involved with the initial modification which installed the P-4/564°F bypass switches. He is familiar with our commitments in ULNRC-03681. He explained to me last night that when ULNRC-03681 was written, the statement of "during startup and shutdown evolutions with Tavg < 564°F just prior to opening the reactor trip breakers" meant we had already inserted rods, we were in MODE 3 and we were now ready to open the reactor trip breakers.

Instead of ULNRC-03681, [redacted] OL Amendment 126 in the 50.59 Screen when stating "the plant can be in MODE 1 or 2, but T-avg must be < 564°F".

OL Amendment 126 supercedes ULNRC-03681 by two years. I do not know what the exact wording in OL Amendment 126 is. If this amendment specifically states (or reasonably implies) "the plant can be in MODE 1 or 2" when bypassing the P-4/564°F FWIS, then it is my belief (I do not wish to speak for [redacted]) that [redacted]s concerns are resolved.

[redacted] CAR was written on the mid-watch when licensing was unavailable for consultation. Tonight, I will attempt to locate and review OL Amendment 126. I believe [redacted] will void his CAR if OL Amendment 126 specifically references MODE 1 or 2.

V/r,
Larry Criscione

From: [redacted]
Sent: Friday, March 30, 2007 2:06 PM
To: Criscione, Larry S.; [redacted].
Cc: [redacted].
Subject: RE: CAR 200703001

Larry and/[redacted]

Call me at home about this tonight.

Thanks,

[redacted]
Ameren UE - Callaway Plant
[redacted]

From: [redacted]
[redacted], March 30, 2007 12:21 PM
To: [redacted]
Cc: [redacted]
[redacted] 200703001
Importance: High

I call this the "Criscione Trail" in honor of the 20+ page email that accompanied this issue until it was answered by my 50.59 screen for OTG-ZZ-00005 Addendum 1 Revision 0. See attached; asked and answered. CAR 200703001 should be voided.

Now maybe I can get back to my hobby as PWROG Licensing Committee chairman and Westinghouse NSSS TSTF rep, since I have to work on those activities on my days off and after hours.

[redacted]

From: Criscione, Larry S.
Sent: Friday, March 30, 2007 4:50 AM
To: DL CAL CARS Screening
Cc [redacted]
Subject: CARS 200703001 - Nuclear Safety Concern with Shutdown

In ULNRC-03681, AmerenUE committed to use the P-4/Lo Tavg FWIS bypass only under the following conditions:

(1) RCS Tavg < 564°F
(2) The plant is in a shutdown evolution and at the point before the Rx trip breakers are opened
(3) The P-4/Lo Tavg FWIS will be restored prior to reaching Mode 2 during a startup evolution.

The concern in CARS 200703001 is that we may not be legalistically meeting our commitment (2) in ULNRC-03681 and literal compliance with T/S 3.3.2 Function 8.a.

It is not clear what "in a shutdown evolution" entails. In his answer to [redacted] CA0139 comment (see attachment to the CARS) [redacted] assumes "in a shutdown evolution" to mean the operators are intending to shutdown the plant. The concern is that when ULNRC-03681 was written, "in a shutdown evolution" meant the rod banks were inserted and the operators were at the point of opening the reactor trip breakers (i.e. the operators were performing the evolution of shutting down the plant and were in MODE 3). Refer to RFR 17015A. If this is how the NRC will interpret "in a shutdown evolution" then they may question whether OTG-ZZ-00005, Addendum 01 meets our commitments to them.

It should be noted that CARS 200703001 does not contend the performance of OTG-ZZ-00005, Addendum 01 is unsafe. The contention is that it may not meet the NRC's interpretation of our ULNRC-03681 commitments.

Based on our current regulatory level of scrutiny, these concerns must be addressed prior to conducting the shutdown. Please ensure CARS 200703001 is screened appropriately.

Thank you,
Larry Criscione

From: [redacted]
Sent: Friday, January 12, 2007 11:56 AM
To: [redacted]
Cc: Criscione, Larry S.; [redacted]
Subject: RE: Re-Screen CAR 200607357

[redacted]

Just wanted to let you know I sent out a message to the design supervisors to let me know if I can make the change from an RFR to an ACNO. As soon as I hear from them, I will be happy to change the CAR type.

[redacted]

From: [redacted]
Sent: [redacted] 2007 6:48 AM
To: [redacted] DL CAL CARS Screening
Subject: RE: [redacted]R 200607357

Then please make the change.
Thanks,

[redacted]

From: [redacted]
Sent: [redacted] 6:41 AM
To: [redacted]
Subject: RE: Re-Screen CAR 200607357

There is no need to send this back to screening as it is not being considered for change from or to an adverse condition. I can make this CARS type change with the concurrence of the appropriate design engineering supervisor.

Thanks,

[redacted]

From: [redacted]
Sent: Friday, [redacted], 2007 6:06 AM
To: DL CAL CARS Screening
Subject: Re-Screen CAR 200607357

Fellow Screeners,
Please re-screen CAR 200607357 from an RFR to an Action Notice. I believe that there is sufficient justification in the string of e-mails below. Larry has provided the following which should be included in the Screening Notes as the justification

> There are no regulatory requirements which need to be addressed to allow using the P-4/564°F FWIS bypass feature in MODE 1 below 564°F, therefore a Request for Resolution is NOT required. The concurrence from Licensing and Accident Analysis of bypassing the P-4/564°F Feed Water Isolation Signal in MODE 1 will be documented in the 10CFR50.59 Screening for OTG-ZZ-00005. RFR 200607357 will remain open as an Action Notice to track this issue to ensure it is specifically documented in the 10CFR50.59 Screening.

Thanks,

[redacted]

From: Criscione, Larry S.
Sent: Thursday, January 11, 2007 9:32 AM
To: [redacted]
Cc: [redacted]
Subject: FW: Tripping the Control Rods into the core During OTG-ZZ-00005

[redacted]

Please have RFR 200607357 re-screened as an Action Notice. The basis is:

> There are no regulatory requirements which need to be addressed to allow using the P-4/564°F FWIS bypass feature in MODE 1 below 564°F, therefore a Request for Resolution is NOT required. The concurrence from Licensing and Accident Analysis of bypassing the P-4/564°F Feed Water Isolation Signal in MODE 1 will be documented in the 10CFR50.59 Screening for OTG-ZZ-00005. RFR 200607357 will remain open as an Action Notice to track this issue to ensure it is specifically documented in the 10CFR50.59 Screening.

If the above reason is not acceptable, then you should be able to find something else from the email trail below.

Larry Criscione

From: [redacted]
Sent: Thursday, January 11, 2007 8:45 AM
To: [redacted]
Subject: RE: RFR 200607357 & OTG-ZZ-00005

The RFR has been reassigned back to Larry.

I'm available to provide whatever support is needed to fill out the CA2510/2511/2512 forms for the procedure change to OTG-ZZ-00005.

[redacted]

-----Original Message-----
From: [redacted]
Sent: Thursday, January 11, 2007 7:03 AM
To: [redacted]
Subject: Fw: RFR 200607357 & OTG-ZZ-00005

[redacted]
See below and please re-assign the RFR.
Thanks,
[redacted]

Sent from my BlackBerry Wireless Handheld

From: Criscione, Larry S.
Sent: Thursday, January 11, 2007 6:28 AM
To: [redacted]
Cc: [redacted]
Subject: RE: RFR 200607357 & OTG-ZZ-00005

[redacted]

That works for me. Please have [redacted] assign RFR 200607357 to me. I will get it re-screened as an Action Notice and use it to track [redacted] suggestion. Based on a discussion I had with [redacted] r, I think there is a good chance we will not pursue this (OTG-ZZ-00005 revision) in cycle 16 – although I know [redacted] intention is to pursue this now. Because of the uncertainty, I would like to capture some of the resolutions from the email trail below in an active Action Notice so the information is available in the event a delay is experienced.

Thanks,
Larry Criscione.

From: [redacted]
Sent: Wednesday, January 10, 2007 6:00 PM
To: Criscione, Larry S.
Cc: [redacted]
Subject: FW: RFR 200607357 & OTG-ZZ-00005

Larry,

Here is one opinion. How would this work for you?

[redacted]

From: [redacted]
Sent: Wednesday, January 10, 2007 2:47 PM
To: [redacted]
Cc: [redacted]
Subject: RE: RFR 200607357 & OTG-ZZ-00005

[redacted]

Based on [redacted] response, I believe that the 10CFR50.59 screening evaluation for the procedure change is the appropriate place to document that this change is within Callaway's current licensing bases. The purpose of 10CFR50.59 is to determine if prior NRC approval is required. It appears that the NRC has appoved this for Callaway back in 1998.

[redacted]

From: Criscione, Larry S.
Sent: Wednesday, January 10, 2007 1:26 PM
To: [redacted] (Callaway Plant)
Cc: [redacted]
Subject: Tripping the Control Rods into the core During OTG-ZZ-00005

[redacted]

I just spoke with [redacted] concerning some research she did earlier in the year on tripping the control rods into the core during a normal shutdown. She informed me she did some benchmarking for you and it is her opinion this is something we should pursue.

I spoke with [redacted] earlier in the day and his understanding is there are still Chemistry concerns regarding this issue.

I first got involved with this issue two years ago when your predecessor [redacted] asked me to pursue it in January 2005. I have gotten nowhere in two years.

Outage Management, Operations and Chemistry need to be aligned on this issue in the next month if it is going to be incorporated into RF16. In January 2006, after reviewing CMP 96-1016A, OL# 1189 and RFR 17015A, Operations concluded the P-4/564°F FWIS could be bypassed in MODE 1. It has taken 12 months to get Licensing to concur with this conclusion. During that 12 months, Operations decided to not include tripping the control rods from MODE 1 in the RF15 scheduling.

Now that we have the "go-ahead" from Licensing, Operations still does not want to incorporate the option of shutting down by tripping the control rods into OTG-ZZ-00005 due to Chemistry concerns.

Operations is currently upgrading the OTGs. If the upgrade of OTG-ZZ-00005 does not include the option of shutting down by tripping the control rods, it is unlikely this option will be available to us in RF16.

What Chemistry concerns exist regarding tripping the control rods during the shutdown? Have the concerns [redacted] posed in the email DTG 200601261129 (several pages below) been addressed?

Thanks,
Larry Criscione

From: [redacted]
Sent: Wednesday, January 10, 2007 11:31 AM
To: Criscione, Larry S.; [redacted]
Subject: RE: RFR 200607357 & OTG-ZZ-00005

Larry,
John will add this option to OTG-ZZ-00005. Thanks for pursuing an RFR to get it on paper.
Rick

From: Criscione, Larry S.
Sent: Wednesday, January 10, 2007 9:48 AM
To: [redacted]
Cc: [redacted]
Subject: RFR 200607357 & Action Notice 200700222

[redacted]

[redacted] requested I send you a CARS on OTG-ZZ-00005. See Action Notice 200700222. [redacted] does not wish to pursue this currently due to outstanding industry concerns (mainly with Chemistry). [redacted]e will be discussing this with you and [redacted].

If we decide not to pursue these changes in the current upgrade of OTG-ZZ-00005 but wish to pursue them during cycle 16, I would suggest assigning Action to Chemistry (and any other concerned departments) to perform the necessary evaluations EARLY in cycle 16 (due dates within 60 days of the end of RF15) to ensure sufficient time is available in cycle 16 for Operations to perform an upgrade.

Larry Criscione

From: [redacted]
Sent: Wednesday, January 10, 2007 9:37 AM
To: [redacted]
Cc: [redacted]
Subject: FW: RFR 200607357 & OTG-ZZ-00005

[redacted],

Please see that we have the proper amount of technical rigor documented in our response to address the issue that Larry describes below. I do not know what type of document is most appropriate, I suspect that an RFR is, but if we have some other licensing or safety analysis way of doing this that is searchable and retrievable, let me know.

Thanks,

[redacted]

From: [redacted]
Sent: Wednesday, January 10, 2007 9:28 AM
To: Criscione, Larry S.
Subject: RE: RFR 200607357 & OTG-ZZ-00005

Come talk to me

[redacted]

From: Criscione, Larry S.
Sent: Wednesday, January 10, 2007 8:44 AM
To: [redacted]
Cc: [redacted]
Subject: RFR 200607357 & OTG-ZZ-00005

[redacted]

In January 2005 (cycle 14), we discussed shutting down the Reactor by tripping the Control Banks into the core. At the time, we agreed we would not make this change for RF14 due to the amount of plant resources needed to address all the other issues associated with RF14. As long as OTG-ZZ-00005 was being upgraded, I wanted to provide the option, regardless of if we would be taking advantage of it in RF14. This was acceptable to you.

OTG-ZZ-00005 was not upgraded during cycle 14. In January 2006 (cycle 15), I discussed shutting the down the Reactor by tripping the Control Banks into the core with [redacted]. RFR 200600140 was written to very specifically address blocking the P-4/564°F FWIS. At the request of Systems Engineering, RFR 200600140 was re-written to be more "general" and then was rejected as not being specific enough to answer. The issue was re-submitted as RFR 200602749 in April 2006 and was rejected by Accident Analysis based on a misunderstanding of what was being requested.

In September 2006, the issue was re-submitted as RFR 200607357. A decision had been made by Operations in August that there was not enough time remaining in the cycle to allow all the necessary issues (major revision to an OTG, possible license amendment, License Operator training concerns) associated with shutting down the Reactor by tripping the Control Banks into the core to be addressed prior to RF15.

The current status of RFR 200607357 is as follows:

- Licensing has stated (see email trail below) no regulatory concerns exist with blocking the P-4/564°F FWIS in MODE 1 as long as Tavg is below 564°F.
- Licensing has stated they are not qualified to answer RFRs
- Accident Analysis has stated that since no concerns exist with blocking the P-4/564°F FWIS prior to tripping the control banks in MODE 1, then RFR 200607357 should be rejected.

I would like to see it documented on a QA record that Accident Analysis and Licensing concur that the P-4/564°F FWIS can be bypassed in MODE 1 below 564°F. RFR 200607357 is the most convenient way to accomplish this. Since no Licensing amendments are necessary, it is my opinion RFR 200607357 can be re-screened as an Action Notice. I want documentation somewhere other than an email in my Inbox – an Action Notice is sufficient for me if that is easier for your groups. If your groups will not provide this documentation, then I can settle for their signatures on a Cross Discipline Review of the procedure – however I cannot speak for the other Operations personnel who will be involved with writing, reviewing and validating OTG-ZZ-00005. Documenting your approval in an Action Notice or RFR will greatly simplify the process of changing the procedure.

[redacted] – Based on my experience with writing, validating and training the crews on OTG-ZZ-00006 in cycle 14, I strongly suggest we move forward with the necessary changes to support shutting down the reactor by tripping the control banks into the core during the upgrade of OTG-ZZ-00005. It is my understanding [redacted] is being retained to upgrade the remaining OTGs. NOTE that just because we have the option of shutting down the reactor by tripping the control banks does not mean we need to do it. IF [redacted] upgrade of OTG-ZZ-00005 is completed before RF15, it need not change any RF15 plans. Procrastinating on this issue during this cycle poses the risk that the necessary changes to OTG-ZZ-00005 will not be made during [redacted] upgrade. If [redacted] contract runs out prior to the OTG-ZZ-00005 changes being incorporated, it is unlikely the resources will be available to properly revise, validate and train on OTG-ZZ-00005 prior to RF16. In cycle 14, we trained on a major revision to OTG-ZZ-00006 while it was still being written and validated. We do not want to repeat that again.

V/r,
Larry Criscione

From [redacted]
Sent: Tuesday, January 09, 2007 9:01 PM
To: Criscione, Larry S.; [redacted]
Subject: RE: RFR 200607357, 200602749 and 200600140

Gentlemen,

In the NRC's Safety Evaluation for LA126 dated 4-23-98, they specifically reviewed and found acceptable our bypass switch design and our using it to block the FWIS initiated by the coincidence of P-4/low T-avg as long as its use was limited to the following plant conditions:

- with T-avg less than or equal to 564°F (you can be in MODE 1 or 2, but must be ≤ 564°F)
- just prior to opening the RTBs (which satisfies the P-4 portion of this feedwater isolation signal's logic).

NRC also wanted this FWIS restored by defeating the bypass prior to entering MODE 2 ascending during startup from an outage. As long as these limitations are observed, no amendment is needed since it's already been reviewed and approved by NRC. We could have been doing this since 4-23-98.

[redacted]

From: Criscione, Larry S.
Sent: Tuesday, January 09, 2007 7:27 AM
To: [redacted]
Cc: [redacted]H.
Subject: RE: RFR 200607357, 200602749 and 200600140

[redacted]

RFR 200607357 requests one of two things be done:

It is requested the Lead for this RFR perform one of the following:

1. Document in the Lead Response to this RFR that there are no regulatory requirements which prevent bypassing the Feed Water Isolation Signal caused by P-4 and Lo Tavg (546°F) prior to tripping the reactor from MODE 1 or 2.
2. Make any necessary amendments to license documents to allow bypassing the P-4/Lo Tavg FWIS prior to tripping the reactor from MODE 1 or 2.

With reference to RFR 200607357, does Bert believe item 1 can be done. I recognize [redacted] does not believe HE can do item 1, but does he believe there is justification for it.

If there is justification for item 1, then RFR 200607357 can be answered by Accident Analysis. If we must go the route of item 2, then RFR 200607357 must be owned by Licensing.

Larry Criscione

From [redacted]
Sent: Tuesday, January 09, 2007 6:41 AM
To: [redacted]
Cc: [redacted] Criscione, Larry S.
Subject: RE: RFR 200607357, 200602749 and 200600140

[redacted] said he could provide a response to Larry's email question, but is not qualified to answer RFRs. If that is required, we will need to get an Engineering person to do it.

[redacted]

[redacted]
Email [redacted]

From: [redacted]
Sent: Monday, January 08, 2007 5:55 PM
To: [redacted]
Subject: RE: RFR 200607357, 200602749 and 200600140

[redacted]
What is our plan?
[redacted]

From: [redacted]
Sent: Thursday, January 04, 2007 3:53 PM
To: [redacted]
Subject: RE: RFR 200607357, 200602749 and 200600140

I plan to talk to [redacted] about this on Monday.

[redacted]

Phone [redacted]
[redacted]
Email [redacted]

From: [redacted]
Sent: Thursday, January 04, 2007 3:32 PM
To: [redacted]
Subject: FW: RFR 200607357, 200602749 and 200600140

[redacted]
Are we on target to address this issue?

Should we have a meeting to ensure we all agree with our direction and what is needed to support each other?
Let me know.
Thanks
[redacted]

From: Criscione, Larry S.
Sent: Wednesday, January 03, 2007 3:13 PM
To: [redacted]
Cc: [redacted]
Subject: RFR 200607357, 200602749 and 200600140

[redacted]

RFR 200607357 is still in Evaluate. It was due on 10/19/2006. The email trail below tells the saga. We have been seeking a resolution since January 9, 2006.

From [redacted] email DTG 200609221821, without any license document changes we can bypass the P-4/564°F FWIS in MODE 1 once we are below 564°F. This should correspond to around 20% reactor power which would be acceptable to me.

I believe RFR 200607357 should be dispositioned as follows:

1. Per item 1 of the RFR, Licensing should affirm we can bypass the P-4/564°F FWIS in MODE 1 once T-avg is below 564°F.
2. Per item 2 of the RFR, Licensing should affirm no licensing amendments are necessary.

From [redacted] email DTG 200609071016 "the time line that needs to be used on this issue" is one that will support implementation in Refuel 16. From my experience with OTG-ZZ-00006 changes in cycle 14, we need an answer prior to RF15 if we are to make major changes to OTG-ZZ-00005 by RF16.

I do not dictate [redacted] priorities. We need consensus between Operations and Licensing on when RFR 200607357 must be answered. My opinion is a due date of no later than March 31, 2007.

How do you want to proceed on this issue?

Larry Criscione

From: Criscione, Larry S.
Sent: Monday, September 25, 2006 6:21 AM
To: [redacted]
Cc: [redacted]
Subject: RE: Tripping the Control Banks in OTG-ZZ-00005; TSTF-444-T

[redacted]

Rick supplied me the responses from the other plants earlier. It does not appear any of them have the capability to block the P-4/Lo Tavg FWIS (this was not part of the original design at Callaway and apparently none of the other plants made the same mod).

We do not need to block P-4/Lo Tavg FWIS to trip the plant. Since we have the capability, it just works better that way:

- Easier on plant equipment – no Aux Feed Water transient thermally shocking the Steam Generators
- Easier on plant operations – no Aux Feed Water transient challenging SG Water Level Control

Being an outlier is not necessarily bad. By definition, if you are in the top tier, you are an outlier. It is my understanding we spent the money on mod package CMP 96-1016A to achieve the two bullets above when opening the trip breakers in MODE 3. Let's take advantage of the fact that at one time these plants were willing to spend money to improve their designs. The mod package was installed years ago; I just want authorization to use it advantageously, now that our operational needs have changed (opening trip breakers in MODE 1 vice MODE 3).

Larry

From: [redacted]
Sent: Friday, September 22, 2006 6:21 PM
To: Criscione, Larry S.
Cc: [redacted]
Subject: RE: Tripping the Control Banks in OTG-ZZ-00005; TSTF-444-T

Larry,

To get the full background behind Commitment (COMN) 43387, you need to read our docketed letter (ULNRC-03681 dated 11/10/97, Att. 1, page 5 of 8) and the NRC Safety Evaluation for LA126 (see attached, Section 2.4, page 2). We told NRC we would use the FWIS bypass switch during startup and shutdown evolutions with T-avg ≤ 564°F just prior to opening the RTBs and we would restore the FWIS prior to entering MODE 2 (during the ensuing startup). In order to abide by that NRC SE, we would not use the bypass switches until T-avg ≤ 564°F which probably doesn't get you very far into MODE 1. We could do that now with no licensing document or COMN changes.

We (Licensing) were told that Operations wants to block FWIS on P-4/low T-avg in order to pursue industry excellence for plants that have short outage durations. However, none of the plants (several plants responded, including Byron and Braidwood that are going to 13 days this fall) that responded to our survey (I've copied you on that separately) said they block this FWIS signal and they trip the reactor in MODE 1. Out of curiosity, why would Callaway need to be an outlier in this area? [redacted] hasn't done anything yet with CAR 200607357, but if it were assigned to me (and I'm sure it will be sometime before it comes due), I'd recommend a footnote be added to Table 3.3.2-1 Function 8a describing the circumstances behind our desire to block this particular enabled function from P-4 (i.e., allow it to be blocked during any of the Applicable MODES for P-4 during a plant shutdown only, to be restored prior to MODE 2 entry ascending). This would not support R-15, but I've arrived at this position based on the following:

- NRC has seen TSTF-444-T via a North Anna amendment. Dominion only used the part of TSTF-444-T that supported their desire to do a P-4 TADOT every 18 months (not after every trip breaker cycle), but North Anna submitted the entire TSTF to NRC in one of their RAI responses. TSTF-444-T needs to be revised generically through the Tech Spec Task Force to discuss blocking specific functions enabled by P-4. We could not block turbine trip on P-4 since it's modeled in several accident analyses, but most of the other P-4 enabled functions could be blocked since they're not credited or modeled in accident analyses. North Anna adopted the Bases portion of TSTF-444-T where only MODES 1 and 2 were listed as being required for turbine trip and FWIS on P-4/low T-avg.

- Although TSTF-444-T uses the Bases to modify the LCO Applicability of the functions enabled by P-4, TSTF-444-T does not say any of those enabled functions can be made inoperable throughout the P-4 Applicability of MODES 1-3.

- Safety Analysis makes a valid point when they say typical ITS rules of usage do not allow the Bases to modify the LCO Applicability.

- There is a program (RITSTF 8a) just getting underway that would try to relocate all RTS and ESFAS permissives out of the Tech Specs, but that program has a 2-year timeline to get NRC approval.

Given our plant-specific desire to block this signal based on the survey responses (no other plant surveyed said they do this), I think the best way to resolve all of the above is to submit a TS change and get a very clear thumbs up or down from NRC. I've copied [redacted] and [redacted] to request that they give me an opinion on the above course of action.

[redacted]

From: Criscione, Larry S.
Sent: Friday, September 22, 2006 1:00 PM
To: [redacted]
Cc: [redacted]
Subject: FW: Tripping the Control Banks in OTG-ZZ-00005

[redacted]

In the attached email you state

> Therefore, I see the desire to use this switch in MODES 1 and 2 as triggering the commitment change process of APA-ZZ-00540 Step 9. The desired change would impact COMN 43387.

The text for COMN 43387 states;

> The administrative controls governing startups will ensure the P-4/Lo-Tavg bypass switch is manually defeated and the isolation function is restored prior to entering Mode 2.

I do not wish to change anything in the above commitment. During the Heat Up/Start Up it is still important that Operations procedures "ensure the P-4/Lo-Tavg bypass switch is manually defeated and the isolation function is restored prior to entering Mode 2." What I wish to do is to bypass the P-4/Lo-Tavg FWIS in MODE 1 or MODE 2 during the shutdown. This does not violate COMN 43387 as written.

What is my next step? Is there a different commitment to which I need to request a change?

Thanks,

Larry

From [redacted].
Sent: Thursday, September 07, 2006 2:30 PM
To: [redacted] Criscione, Larry S.
Subject: RE: Tripping the Control Banks in OTG-ZZ-00005

There you go Larry, begin with APA-ZZ-00540

From: [redacted]
Sent: Thursday, September 07, 2006 10:51 AM
To: Criscione, Larry S.
Cc: [redacted]
Subject: RE: Tripping the Control Banks in OTG-ZZ-00005

Larry,

The Safety Analysis group is not opposed to a TS Bases change. Our point was that we interpreted the TS such that a TS change would be required. However, we deferred to Licensing as the TS Subject Matter Experts. If [redacted] believes and can document why a TS change is not needed and that only a TS Bases change is sufficient, we have no strong objections.

[redacted]

From: Criscione, Larry S.
Sent: Thursday, September 07, 2006 10:24 AM
To: [redacted]
Cc: [redacted]
Subject: RE: Tripping the Control Banks in OTG-ZZ-00005

In order to meet a Refuel 16 timeline, all license amendments must be implemented prior to issuing OTG-ZZ-00005 for training in cycle 16. What is an acceptable timeline for evaluating (i.e. deciding a course of action, assigning Actions for evaluating and implementing license changes, and taking the RFR to InProcess) RFR 200607357?

My experience is that if we do not set earlier milestones, we will be pressing people at the last minute to meet later milestones. I was unaware a final decision for RF15 had been made prior to requesting an October 6 due date for RFR 200607357. Regardless, I think 30 days is sufficient time to evaluate this issue and set out a plan for RF16.

From: [redacted]
Sent: Thursday, September 07, 2006 10:16 AM
To: Criscione, Larry S.; [redacted]
Cc: [redacted]
Subject: RE: Tripping the Control Banks in OTG-ZZ-00005

This will not happen for Refuel 15. The Operations Refuel meeting August 21 decided to pursue this for Refuel 16. That is the time line that needs to be used on this issue.

[redacted]

From: Criscione, Larry S.
Sent: Thursday, September 07, 2006 9:09 AM
To: [redacted]
Cc: [redacted]
Subject: FW: Tripping the Control Banks in OTG-ZZ-00005

[redacted],

Below is the email trail on the RFR we discussed this morning. During our discussion I stated the plan was to remove P-4 from T/S via the WOG. This is a stand alone issue regardless of RFR 200607357. It is related to the RFR in that once it is done, there will be no question as to whether the P-4/564°F FWIS can be bypassed at power.

From [redacted]s email (two below) this morning, it appears RFR 200607357 can be resolved with a T/S Bases change. This could occur prior to RF15 if we want to drive it. This would allow us to trip the reactor after taking the turbine off-line without any feed water transient. Is this something we want to pursue for the RF15 shutdown?

Larry

From: Criscione, Larry S.
Sent: Thursday, September 07, 2006 8:59 AM
To: [redacted].
Cc: [redacted]
Subject: RE: Tripping the Control Banks in OTG-ZZ-00005

[redacted],

The red highlighting in [redacted] response was added by me. I think it is the crux of the matter. Do you disagree with it or know of any accident analysis which relies on a post trip FWIS below 546°F? Can you support a T/S Bases change?

Larry

From: [redacted]
Sent: Thursday, September 07, 2006 8:54 AM
To: Criscione, Larry S.; [redacted]
Cc: [redacted]
Subject: RE: Tripping the Control Banks in OTG-ZZ-00005

I haven't seen anything on this commitment change nor has [redacted] talked to me about it. Following APA-ZZ-00540 is the first step in the process of changing a commitment and evaluating whether prior NRC approval is required. Depending on that outcome, we would then determine whether a Bases-only or TS change is required. I maintain this can be done via a Bases change only; P-4 is operable whenever it receives the required RTB position inputs and the permissive is satisfying its logic outputs to SSPS. The enabled function that relies on P-4, FWIS on low T-avg, is not required by ESFAS Function 5 nor credited in any accident analysis that I'm aware of.

[redacted]

From: Criscione, Larry S.
Sent: Thursday, September 07, 2006 8:44 AM
To: [redacted]
Cc [redacted]
Subject: RE: Tripping the Control Banks in OTG-ZZ-00005

I was told in January that [redacted] was doing the APA-ZZ-00540 paperwork. Is this is something I need to do, please let me know.

From: [redacted]
Sent: Thursday, September 07, 2006 8:38 AM
To: [redacted]
Cc: [redacted].; Criscione, Larry S.
Subject: RE: Tripping the Control Banks in OTG-ZZ-00005

Yes.

I sent [redacted] all of the plant survey responses that I received on blocking this FWIS function. All plants that trip the reactor in MODE 1 deal with the FWIS, reset it and restore normal feed as quick as they can. None block the FWIS on P-4/low T-avg or have the block switch design that we installed. Safety Analysis is the group that provided the less-than-adequate RFR response, not Engineering. I told [redacted] that [redacted] STF to revise TS 3.3.2 (and change TSTF-444-T) would take at least 2 years dealing with the TSTF process. [redacted] and his group don't support doing just a Bases change. At the last WOG LSC meeting, [redacted]) said he thought we could pursue just a Bases change, but we can't get that through ORC unless [redacted] is on board. I could do a Bases change if you get [redacted] to support it.

I don't think this RFR should be assigned to Licensing. The answer to the first question is "Yes." We made this commitment to NRC in a docketed letter (ULNRCs are part of our licensing basis) during the OL1169 review and no one that wants to change that commitment has followed APA-ZZ-00540 yet. APA-ZZ procedures apply to everyone. The 2nd question deals with ODs that might credit this function and we don't write ODs or even know about all the ODs that may have been written.

[redacted]

From: [redacted]
Sent: [redacted] September 06, 2006 7:34 AM
To: [redacted]
Subject: FW: Tripping the Control Banks in OTG-ZZ-00005

[redacted] is this a continuation of the exchanges you have had with [redacted]? Let's discuss this and where you think it is going.

[redacted]

From: [redacted]
Sent: [redacted] September 05, 2006 2:07 PM
To: [redacted]
Subject: FW: Tripping the Control Banks in OTG-ZZ-00005

From: Criscione, Larry S.
Sent: Tuesday, September 05, 2006 1:28 PM
To: [redacted]
Cc: [redacted]
Subject: FW: Tripping the Control Banks in OTG-ZZ-00005

[redacted]

Please send RFR 200607357 to [redacted] and not [redacted]. This is a licensing issue and, as demonstrated by the response to RFR 200602749, Engineering is not willing to address it.

> We have been seeking an answer to this issue since January 9. RFR 200602749 was originally submitted as RFR 200600140 and then reworded at the request of Engineering. The new, vaguely worded RFR 200600140 was then rejected. RFR 200602749 was then submitted with the original wording of RFR 200600140. RFR 200602749 was closed without addressing the question. From the attached email [redacted] and [redacted] appear to be prepared to answer this issue. The History List on RFR 200602749 indicates it was sent to Accident Analysis and not Licensing.

OTG-ZZ-00005 must be upgraded by October 31, 2006 to meet a Refueling Milestone. We cannot afford to avoid answering this issue any longer. Thirty days may not be much time to evaluate this issue, but in Operations we have been awaiting an answer for nine months. If [redacted] is not the right individual for this RFR, please ensure it is appropriately assigned at Screening tomorrow to an individual who can provide an evaluation by October 6, 2006. Note we do not need licensing amendments by October 6; we only need an evaluation as to what license amendments (if any) must be initiated and their feasibility.

Thank you for your assistance.

Larry Criscione

From: [redacted]
Sent: [redacted] ember 05, 2006 11:26 AM
To: Criscione, Larry S.
Subject: RE: Tripping the Control Banks in OTG-ZZ-00005

Larry,
Please write a new RFR to [redacted].
[redacted]

From: Criscione, Larry S.
Sent: Mon[redacted]ptember 04, 2006 3:56 PM
To: [redacted]
Cc: [redacted]
Subject: Tripping the Control Banks in OTG-ZZ-00005

[redacted]

Do you still want the option to trip the control banks in OTG-ZZ-00005?

> RFR 200602749, *Evaluate Bypassing P-4/564°F FWIS during MODE 1 (15%) or 2 Manual Trip*, was not answered correctly. The two questions posed still remain unanswered:
>
> 1. Do current licensing documents require the P-4/Lo Tavg FWIS to be operable in MODE 1 or 2 (the originator of this RFR could not find any requirement)?
> 2. Do any Operability Determinations rely on the P-4/Lo Tavg FWIS being available?
>
> The Lead Response to the RFR answered the RFR by evaluating if the P-4 function is required. **We do not wish to disable the P-4 function**; we only want to bypass the **FWIS** which is caused by the simultaneous inputs of P-4 and temperature less than or equal to 564°F.

The last word I got was we are no longer considering tripping the control banks due to chemistry concerns, so I have chosen not to pursue an appropriate closure of RFR 200602749.

Please let me know if I need to continue to pursue this.

Thanks,

Larry Criscione

From: [redacted]
Sent: [redacted]er 22, 2006 5:45 PM
To: Criscione, Larry S.
Subject: FW: survey on feedwater isolation from P-4 and low T-avg

FYI – see plant survey responses.

[redacted]

From: [redacted]
Sent: [redacted] 22, 2006 11:40 AM
To: [redacted]
Subject: RE: survey on feedwater isolation from P-4 and low T-avg

Yes.

[redacted]

From: [redacted]
Sent: [redacted]ust 22, 2006 11:39 AM
To: [redacted]
Subject: RE: survey on feedwater isolation from P-4 and low T-avg

[redacted]
Does all that equal 2 years from today?
[redacted]

From: [redacted]
Sent: [redacted] 22, 2006 11:24 AM
To: [redacted]
Cc: [redacted]
Subject: RE: survey on feedwater isolation from P-4 and low T-avg

[redacted]

Any T/S change we pursue would need to be coordinated with a revision to TSTF-444-T to reflect the NRC-approved North Anna amendment and this TSTF revision must first be processed through the PWROG. E-bar elimination took 5 years to work its way through the PWROG process from initial conception to final TSTF-490 submittal to NRC (2001-2005) and our OL#1257 amendment submittal (5-9-06). This shouldn't take as long as E-bar took, but I would guess 12 months to get the PWROG (which now includes CE and B&W) to agree on a revised TSTF-444-T. I will work on a Callaway-specific amendment in parallel with the PWROG, but NRC would likely take a year to review a submittal if it came in today and we are a ways off from making an amendment submittal.

[redacted]

From: [redacted]
Sent: [redacted], August 22, 2006 9:21 AM
To: [redacted]
Subject: FW: survey on feedwater isolation from P-4 and low T-avg

[redacted]
Can you pursue the T/S change? Based on your response it will take some time and probably will not happen in time for RF 15. Do you have a estimated completion date?
[redacted]

From: [redacted]
Sent: Monday, August 21, 2006 1:33 PM
To: [redacted]
Subject: RE: survey on feedwater isolation from P-4 and low T-avg

[redacted]

I discussed this with [redacted], and it appears based upon our previous discussions that Licensing [redacted] would be the best since this issue deals with the Tech. Specs./Tech. Spec. Bases and applicabiliy of the functions rather than what is credited in the Safety Analysis. We are willing to say, as discussed in the CARS/RFR 200602749, this function is not credited for accident mitigation in the Safety Analysis. Safety Analysis can support this by performing a CDR review of the change, as desired.

[redacted]

From: [redacted]
Sent: Monday, August 21, 2006 12:55 PM
To: [redacted]
Cc: [redacted]
Subject: FW: survey on feedwater isolation from P-4 and low T-avg

[redacted]

Is your group willing to support a T/S Bases change? The drop dead date for RF 15 is 11/17/06.

[redacted]

From: [redacted]
Sent: Tuesday, August 15, 2006 10:26 AM
To: [redacted]
Cc: [redacted]
Subject: RE: survey on feedwater isolation from P-4 and low T-avg

[redacted]

We discussed TSTF-444-T and P-4 during the PWROG TS Working Group meeting on 8/1 and 8/2. I don't recommend adopting TSTF-444-T, as written, besides NRC would not approve a license amendment by the date it would be needed for next March's outage. However, all of Working Group meeting attendees were in agreement (at least I heard no spoken disagreement) that a TS Bases change could be pursued under 50.59 to accomplish what you want. Our MODE 3 restriction on using the
P-4/low T-avg block switch is captured under COMN 43387, but this commitment is not captured in the TS or Bases. I recommend that you check with [redacted] and [redacted] to see what is required to revise a commitment under APA-ZZ-00540 and we start the process of evaluating whether prior NRC notification of a commitment change is needed. I heard no disagreements during the Working Group meeting when I said that I believe P-4 is operable as long as the permissive is receiving inputs from the breaker position switches and the permissive outputs are providing the required logic inputs to the SSPS to enable the various end device functions. I said that I don't believe there should be any requirement for the 3.3.2 Bases to discuss non-credited, end device functions. Our TS Bases (pages B 3.3.2-36 and -37) could be modified to make this type of statement and to qualify the text on the enabled functions to either delete them or state that their functional status, even if not functional, does not render the permissive itself inoperable. The discussion of what P-4 enables could also be revised to delete any mention of steam dump arming, clearly not a safety analysis required end device function. I will be following this issue as it develops through the process at the PWROG Licensing Subcommittee (LSC) and TS Working Group, but that process takes time and would not support a short term TS Bases change. I told [redacted] that if the Safety Analysis group is wary of making a Bases change at this time, in advance of the LSC and TS Working Group resolution of the issue, [redacted] and [redacted] should convey that position to Operations. A longer term project authorization PA-LSC-0331 on RITSTF 8a was approved at the meeting and would call for

Westinghouse to develop an argument that none of the RTS or ESFAS permissives satisfy any of the 4 criteria under 50.36 for TS inclusion and justify their relocation from the TS – however that program has a 2-year timeline.

Here's what I've received back so far on the survey responses. Most people live with the FWIS when they trip at power. I also heard that North Anna received NRC approval for a portion of TSTF-444-T. They cited TSTF-444-T to justify an 18-month TADOT frequency for P-4 rather than the STS NUREG-1431 TADOT frequency of checking P-4 after every RTB cycle. Their P-4 Applicability remains Modes 1-3 in TS Table 3.3.2-1, but they have varying applicability requirements in the Bases for the functions enabled by P-4 (their NRC-approved Bases say FWIS on P-4 and low T-avg is required only in Modes 1 and 2).

[redacted]

Here is the Braidwood/Byron response. If you need additional information you can contact [redacted]. He provided the response or give me a call at [redacted].

1. Does your plant design include FWIS on P-4 coincident with low RCS T-avg?

Yes

2. If yes, do you allow normal feedwater to be isolated when RCS T-avg reaches its setpoint (564°F at Callaway)? If so, does your Operations staff share the above concerns expressed at Callaway?

No. We generally trip the reactor from 8 to 10 % power after the turbine has been tripped. There is usually a 5 minute delay between turbine trip and reactor trip which allows us to stabilize a bit. FW temp is about 275 or so at the time of the reactor trip. We expeditiously reset the FW isolation and establish low flow feed. We have a startup FW pump so AF is not used.

3. If you block FWIS on P-4 with low T-avg during shutdown, how would reconcile that against the TS Table 3.3.2-1 requirement that P-4 (and the TS Bases inference that enabled functions from P-4) be Operable in Modes 1-3?

We do not block P-4.

[redacted]

[redacted]

-----Original Message-----
From: [redacted]
To: [redacted]
Sent: Wed Aug 09 14:44:15 2006
Subject: RE: survey on feedwater isolation from P-4 and low T-avg

Responses:

Q1: Yes, same setpoints as at Callaway.

Q2: Yes, Vogtle verifies proper FWI in the controlling procedure (Vogtle would not enter EOP E-0). Currently we have not, to my knowledge, experienced any significant increase in dose rates.

Q3: No, we do not block FWI.

One thing I noticed in the benchmarking data is the Rx power at which VEGP would initiate a manual trip for the shutdown is stated as "5%". This is low. We would initiate the trip around 18% - 20% Rx power.

[redacted]

[redacted]-Original Message-----
From: [redacted]
Sent: [redacted], August 09, 2006 3:08 PM
To: [redacted]
Subject: Fw: survey on feedwater isolation from P-4 and low T-avg

[redacted]

Could you take a look at [redacted] questions below?

Thanks,

[redacted]

[redacted]

Answers inserted below.

[redacted]

Please answer the following three questions:

1. Does your plant design include FWIS on P-4 coincient with low RCS T-avg? Yes

2. If yes, do you allow normal feedwater to be isolated when RCS T-avg reaches its setpoint (564°F at Callaway)? Yes. If so, does your Operations staff share the above concerns expressed at Callaway? NO. CPSES has proceduralized the actions required to control the resulting secondary system transient (i.e., minimize the potential for lifting FWH reliefs, prevent tripping of MFP's, auto start of AFW, etc.). For more info re: procedures contact [redacted]

3. If you block FWIS on P-4 with low T-avg during shutdown, how would reconcile that against the TS Table 3.3.2-1 requirement that P-4 (and the TS Bases inference that enabled functions from P-4) be Operable in Modes 1-3? CPSES design does not include block capability for FWIS.

[redacted] see responses below. Seems this condition has been discussed by engineering recently and may be the subject of future discussions regarding the acceptability of tripping and creating this FWI.

Thanks,

[redacted]

Sequoyah Nuclear Plant

[redacted]

Please answer the following three questions:

1. Does your plant design include FWIS on P-4 coincient with low RCS T-avg?

   Yes

2. If yes, do you allow normal feedwater to be isolated when RCS T-avg reaches its setpoint (564°F at Callaway)? If so, does your Operations staff share the above concerns expressed at Callaway?

   Yes. Not identified by operations at this time. However, system engineering has noted a higher than usual maintenance need for FW relief valves and the pressure seen during these trips could be the cause.

3. If you block FWIS on P-4 with low T-avg during shutdown, how would reconcile that against the TS Table 3.3.2-1 requirement that P-4 (and the TS Bases inference that enabled functions from P-4) be Operable in Modes 1-3?

   N/A

From: [redacted]
Sent: [redacted]ust 14, 2006 2:17 PM
To: [redacted]
Subject: RE: survey on feedwater isolation from P-4 and low T-avg

[redacted] you received any feed back?

[redacted]

From: [redacted]
Sent: [redacted]
Subject: survey on feedwater isolation from P-4 and low T-avg

Callaway Plant would like your help responding to a short survey. I brought up an issue at last week's PWROG Licensing Subcommittee / TS Working Group meeting regarding our desire to manually trip the reactor during MODE 1 in our March 2007 refueling outage. The following information provides some background on what our Operations staff would like to do:

Callaway would like to incorporate a planned "hard shutdown" which means a manual reactor trip after the generator breakers are open and the turbine is secured. The reason behind this initiative is to save approximately 3-4 hours of critical path time and align Callaway with the industry excellent plants in the area of refuel outage duration. The concern with the feedwater isolation signal (FWIS) that occurs on the coincidence of P-4 and low T-avg (564°F at Callaway) is the disruption of feedwater from the normal feedwater system which would cause an overpressure situation due to feed pre-heating, resulting in relief valves lifting, trip of the main feed pumps, and the potential need to use AFW for inventory control.

The following are the initial conditions:
Reactor Power – 10 to 20 %
Main Generator Breakers open
Turbine tripped .
Condenser Steam Dump System armed in Steam Pressure Mode maintaining RCS temperature at approximately 560 F
RCS pressure at 2235 psig

At last week's meeting it became clear that no plant had adopted TSTF-444-T which requires P-4 to be Operable in Modes 1-4, but has differing Applicability requirements in the Bases for enabled functions (FWIS on P-4 and low T-avg required in Modes 1 and 2 per TSTF-444-T). Based on benchmarking data given to me (see attached files), Braidwood, Byron, Cook, McGuire, Vogtle, Farley (last outage), Wolf Creek, Comanche Peak, Sequoyah, Watts Bar, Salem, Beaver Valley, Calvert Cliffs, Ginna, and Palo Verde trip the reactor in Mode 1 (varying between 6% and 25% RTP) during refueling outage shutdowns.

Please answer the following three questions:

1. Does your plant design include FWIS on P-4 coincient with low RCS T-avg?
2. If yes, do you allow normal feedwater to be isolated when RCS T-avg reaches its setpoint (564°F at Callaway)? If so, does your Operations staff share the above concerns expressed at Callaway?
3. If you block FWIS on P-4 with low T-avg during shutdown, how would reconcile that against the TS Table 3.3.2-1 requirement that P-4 (and the TS Bases inference that enabled functions from P-4) be Operable in Modes 1-3?

Thanks,

[redacted]

From: [redacted]
Sent: Thursday, May 25, 2006 9:47 PM
To: Criscione, Larry S.
Cc: [redacted]
Subject: RE: RFR 200600140 & RFR 200602749

[redacted]
We are not requesting to block P-4. Our request is to disable one of the many functions of P-4.
[redacted]

From: Criscione, Larry S.
Sent: Thursday, May 25, 2006 7:27 AM
To: [redacted]
Cc: [redacted]
Subject: RFR 200600140 & RFR 200602749

[redacted]

Please add the following to RFR 200600140:

Identify and initiate required FSAR and Tech Spec changes to allow blocking the P-4/564°F Feed Water Isolation Signal in MODE 1 with the turbine offline.

See RFR 200602749.

Thanks,
Larry Criscione

From: [redacted]
Sent: Thursday, January 26, 2006 12:25 PM
To: Criscione, Larry S.
Cc: [redacted]
Subject: FW: OL#1189 FWIS bypass switch use in MODES 1 and 2

Larry,
Could you give [redacted] more information of how the shutdown would occur? Aren't we still going to conduct a plant shutdown to below the swap-over of the feed reg valves and get to essentially 0% power before we trip? Understand Operations is revising OTG-ZZ-00005 to permit opening the Reactor Trip breakers in Mode 1 after removing the main generator from service from your RFR 200600140.
I have [redacted] looking into the commitment change process that [redacted] has advised me about.
[redacted]

-----Original Message-----
From: [redacted]
Sent: Thursday, January 26, 2006 11:29 AM
To: [redacted]
Cc: [redacted]
Subject: RE: OL#1189 FWIS bypass switch use in MODES 1 and 2

[redacted]

It was not clear to me what you were asking for in the request below. It is my understanding that Outages is looking into the possibility of tripping the plant from approximately 30% power. I would like to provide some justification for **not** doing this so that questions can be asked when benchmarking other plants that have experience in tripping their plant during shutdown. I know STP has recently experienced several refuelings with very large particulate releases that have caused higher than predicted dose rates. One reason for this particulate release may be due to running all 4 RCPs (high flow rates may be causing the crud release). The shutdown data for Co-58, Co-60, Cr-51 nickel and iron data should be collected for the entire shutdown cleanup (both particulate and soluble data). This should include data from before the trip and shutdown. The plant's history of dose rate and contamination trends should also be reviewed.

From a chemistry cleanup and dose prospective, the best type of shutdown is a smooth one (with no abrupt changes, trips, water hammers, rod drop tests, etc. that could cause a large particulate crud release). Operating history shows that plants performing rod drop tests during shutdown have experienced large particulate crud releases early during the shutdown (with particulates dropping out in out of core areas causing high dose rates). These same plants have experienced high activity particulates during steam generator tube inspections causing high dose and contamination issues. Since Callaway will be completing the baseline steam generator inspections during RF15, it would not be a good refuel to have high dose rates, nor do we want to deposit high activity crud in our new steam generators during the first shutdown following replacement.

Chemistry has already estimated the peak Co-58 release for RF15 and it is expected to be the largest release for Callaway due to the SGR (increased release rates for the first few cycles of operation and 45% increase in tubing surface area). However, this release should be soluble and easy to cleanup. Actually by injecting zinc during the cycle, we expect our peak Co-58 to be reduced by a factor of 2-3. The RF15 estimated cleanup time required prior to stopping the RCPs is estimated to be 33 hours. For Refuel 16 and beyond (after the oxide layer is formed) this peak should be reduced considerably with almost no cleanup time required (just a few hours).

Recent information presented by EPRI indicates that it is no longer critical to cool down slowly or to stay in acid-reducing conditions for 12 hours during shutdown so there may be other ways to reduce the time to offload fuel. I realize that short refueling are the best thing for Callaway (I'm all for <20 days) but we need to be careful and consider all the consequences.

[redacted]

[redacted]
Consulting Chemist
AmerenUE
[redacted]

-----Original Message-----
From: [redacted]
Sent: Wednesday, January 25, 2006 3:38 PM
To: [redacted]
Cc: [redacted]
Subject: OL#1189 FWIS bypass switch use in MODES 1 and 2

[redacted]

In response to your question on allowing the use of the feedwater isolation bypass switch in MODE 1 or MODE 2, I uncovered the following. We categorized the bypass switch design change as an Unreviewed Safety Question in OL#1189 (ULNRC-03628 dated 8/8/97) under the old 50.59 rule and limited its use to prior to MODE 2 entry in ULNRC-03681 dated 11/10/97. NRC's Safety Evaluation for Amendment 126 dated (4/23/98 (bottom of page 2) also limited the use of the bypass switch to "prior to entering MODE 2." However, nothing was added to the Tech Specs, the Tech Spec Bases, or the FSAR on this MODE 2 limitation and current Bases page B 3.3.2-37 has no such restriction.

Therefore, I see the desire to use this switch in MODES 1 and 2 as triggering the commitment change process of APA-ZZ-00540 Step 9. The desired change would impact COMN 43387. I am not involved with the Commitment Tracking Process, nor have I ever been, but reading through the procedure I would advise the following:

1. Person requesting the change fills out forms CA1571 and CA2358 per step 9 of APA-ZZ-00540.
2. Forms should be sent to [redacted] (as of 2/1/06, Commitment Tracking transfers from Mark Reidmeyer's old RRA group to [redacted] group [redacted] and [redacted]).
3. Most of the CA2358 form questions will be answered "No" based on the discussion in the OL#1189 submittal (ULNRC-03628 discusses how FWIS on P-4 with coincident low T-avg is not credited in any accident analyses); however, question 4 on the form needs to be answered "Yes" since the commitment is in an NRC SE. I think this change will be allowed, but must be described in the "next commitment update report", whatever that is. Maybe MAR can provide more background, but Gail or Pat will be the ones actually responsible for this report.

I have no further knowledge of the commitment change process, but the above should get the process started.

[redacted]

From: [redacted]
Sent: Wednesday, January 11, 2006 8:22 AM
To: Criscione, Larry S.
Subject: RE: RFR 200600140

Larry,
Based on this e-mail, I have taken this CAR back to Initiate for you.

Thanks,
[redacted]

-----Original Message-----
From: Criscione, Larry S.
Sent: Wednesday, January 11, 2006 8:14 AM
[redacted]
Cc: [redacted]
Subject: RE: RFR 200600140

That works for me.

-----Original Message-----
From: [redacted]
Sent: Wednesday, January 11, 2006 7:35 AM
To: Criscione, Larry S.
Subject: RE: RFR 200600140

Larry, [redacted] talked to me and said he had discussed with you about rewritting the request for this RFR. If that is correct, I will reassign it to you and you can have [redacted] (or someone else) take it back to initate, so you can redo the write up.

[redacted]

> -----Original Message-----
> **From:** Criscione, Larry S.
> **Sent:** Tuesday, January 10, 2006 10:59 AM
> **To:** [redacted]
> **Cc:** [redacted]
> **Subject:** RFR 200600140
>
> [redacted],
>
> [redacted] volunteered to answer RFR 200600140 yesterday since he was involved with the original Mod Package. I need a preliminary answer soon. I intend to work on this with Reactor Operator and training department resources in the simulator in February. I do not want to waste their time down a wrong path.
>
> I did not give a due date on the RFR since I do not need a formal written answer until early next year. However, I need someone to look at the issue this month to let me know definitively if the P-4/564°F FWIS can be bypassed in Mode 1.
>
> Thanks,
> Larry

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D C 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO 125 TO FACILITY OPERATING LICENSE NO NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO 50-483

1 0 INTRODUCTION

By letter dated August 8, 1997, as supplemented by letter dated November 10, 1997, the Union Electric Company (UE) requested changes to the Technical Specifications (Appendix A to Facility Operating License No NPF-30) for the Callaway Plant, Unit 1 The proposed changes would revise the Technical Specifications (TS) to change feedwater isolation engineered safety features actuation system (ESFAS) functions in TS Tables 3 3-3, 3 3-4, and 4 3-2

The November 10, 1997, supplemental letter provided additional clarifying information and did not change the staff's original no significant hazards determination that was published in the Federal Register on December 17, 1997 (62 FR 66144)

2 0 TECHNICAL SPECIFICATION CHANGES AND EVALUATION

2 1 Actuation Logic Applicability

The applicable modes for Functional Units 5 a 1), Automatic Actuation Logic and Actuation Relays (SSPS), and 5 a 2), Automatic Actuation Logic and Actuation Relays (MSFIS), in Tables 3 3-3 and 4 3-2 would be revised to add MODE 3 This change is proposed because the automatic actuation logic for closure of the main feedwater isolation valves (MFIVs) must be available in MODE 3 to establish a pressure boundary preventing diversion of auxiliary feedwater (AFW) flow, thereby ensuring delivery of AFW flow to at least two intact steam generators under accident conditions As a result of this change in applicability, the end point of the action statements will be changed to hot shutdown This change is more restrictive and is consistent with the applicability of other TS related to decay heat removal by the auxiliary feedwater (AFW) system This change is acceptable

2 2 New Steam Generator Level Low-Low Functional Unit

A new Functional Unit 5 d, Steam Generator (SG) Water Level Low-Low (for feedwater isolation only), would be added to Tables 3 3-3, 3 3-4, and 4 3-2 This change is more restrictive The main feedwater isolation valve (MFIV) isolation on SG water level low-low isolation was added to the plant design to address a concern that AFW flow could be fed back through the MFW system instead of to the SGs under certain break conditions This isolation signal is credited in the analyses for the loss of non-emergency AC power, loss of normal feedwater, and feedwater system pipe break events This isolation signal was not included in the original TS, which were based on the Westinghouse Standard Technical Specifications (STS), because neither the STS at the time nor the current STS include this isolation signal While this isolation signal had not previously been included in the TS, the licensee stated that they have always performed surveillances on this isolation signal consistent with other automatic actuation logic and actuation relays applicable in MODES 1-3 This change is acceptable

2 3 Trip Time Delay Applicability

The applicable MODES in Table 3 3-3 for auxiliary feedwater (AFW) SG Water Level Low-Low Functional Units 6 d 1) c), Start Motor Driven Pumps Vessel Delta T (Power-1, Power-2), and 6 d 2) c), Start Turbine-Driven Pump Vessel Delta T (Power-1, Power-2), would be revised to delete MODE 3 Functional Unit 6 d 3) in Table 4 3-2 would also be revised to delete MODE 3 This function is used to change the trip time delays depending on power level At reactor thermal power less than or equal to 10 percent, the maximum trip time delay is enabled, and the maximum trip time delay should always be enabled in MODE 3 This change is acceptable

2 4 Feedwater Isolation on P-4/Low Tavg

The Bases for Functional Unit 11 b, Reactor Trip P-4, in Table 3 3-3 would be revised to add a note allowing the feedwater isolation function on P-4 (reactor trip and bypass breakers open) coincident with low Tavg (Tavg $\leq$ 564°F) to be blocked The reason for the change is to decrease unnecessary cycling of the MFIVs and AFW system which adversely impacts startup and shutdown evolutions This feedwater isolation function provides backup protection for excessive cooldown events and is not credited in any FSAR analyses The licensee has proposed to install a bypass switch to block this signal during startup and shutdown evolutions with Tavg $\leq$ 564°F just prior to opening the reactor trip breakers The feedwater isolation function would be restored by manually defeating the bypass prior to entering MODE 2 This change is acceptable

2 5 <u>Conclusion</u>

The staff has reviewed the licensee's proposed TS changes to revise the feedwater isolation ESFAS functions Based on the review, the staff concludes that the proposed TS changes are acceptable

3 0 STATE CONSULTATION

In accordance with the Commission's regulations, the Missouri State official was notified of the proposed issuance of the amendment The State official had no comments

4 0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 66144) Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51 22(c)(9) Pursuant to 10 CFR 51 22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment

5 0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public

Principal Contributor A Cubbage

Date April 23, 1998

50.59 SCREEN

I. **Activity/Document Number:** OTG-ZZ-00005 Addendum 1 **Revision Number:** 0

Title: Opening Reactor Trip Breakers in MODE 2 - ITPE

Brief Description of activity (what is being changed and why):
New Addendum being issued to provide instructions for opening the reactor trip breakers (RTBs) in MODE 2.

Continued ☐

II. **Applicability Determination**
Other applicable processes identified during the applicability determination: No other processes identified.

(If a CA2510 was NOT filed, indicate that no other processes apply or document the basis for not performing a CA2510).

III. List the documents (FSAR, Technical Specifications, Technical Specification Bases, and other documents) reviewed, including section numbers, where relevant information was found. (Relevant documents listed in the parent document (RFR, Procedure, etc. need not be repeated).):

OL Amendment 126 dated 42398; ULNRC-03681 dated 11-10-97; ULNRC-03628 dated 8-8-97; COMN 43387; CAR 200700230; CAR 200700222; CAR 200607357; CAR 200606241; CAR 200506711

Continued ☐

IV. Identify the relevant FSAR-described SSC(s) and the associated design function(s) (See Sections 3.3 and 5.1.1 of the 10CFR50.59 Resource Manual (RM)):

Reactor Trip System (TS 3.3.1, FSAR Section 7.2); Engineered Safety Features Actuation System (TS 3.3.2; FSAR Sections 7.3.7 and 7.3.8) – design functions include tripping the reactor and isolating an automatic feedwater isolation signal

Continued ☐

V. **50.59 Screening Questions:**
(Check the correct responses below. Attach additional page(s) to provide justification for "YES" response(s) if desired. See Section 5.2.2 of RM for additional guidance.)

1. Does the proposed activity involve a change to an SSC such that it adversely affects an FSAR-described design function? (See Section 5.2.2.1 of the RM.) ☐ YES ☒ NO
2. Does the proposed activity involve a change to a procedure such that it adversely affects how FSAR- described SSC design functions are performed or controlled? (See Section 5.2.2.2 of the RM.) ☐ YES ☒ NO
3. Does the proposed activity involve revising or replacing an FSAR-described evaluation methodology that is used to establish the design bases or used in the safety analyses? (See Section 5.2.2.3 of the RM.) ☐ YES ☒ NO
4. Does the proposed activity involve a test or experiment not described in the FSAR, where an SSC is utilized or controlled in a manner that is outside the reference bounds of the design for that SSC or is inconsistent with analyses or descriptions in the FSAR? (See Section 5.2.2.4 of the RM.) ☐ YES ☒ NO
5. Does the proposed activity involve or require a change to the Technical Specifications? * (See Section 5.2.2.5 of the RM.) ☐ YES ☒ NO

VI. If all questions are answered NO, then implement the activity per the applicable plant procedure for the type of activity without obtaining a License Amendment.

If screen question 5 is answered yes, then request and receive a License Amendment prior to implementation of the activity. *

If screen question 1, 2, 3 or 4 is answered YES, then a 50.59 Evaluation shall be performed and approved prior to implementation of the activity.

* A 10CFR50.59 evaluation is NOT required for FSAR and Technical Specification (TS) Bases change made strictly to conform the FSAR and TS Bases to Technical Specification changes approved by the NRC. Such changes are typically implemented in conjunction with implementation of the NRC-approved TS changes without further evaluation.

VII. If the conclusion of the screening questions is that a 50.59 Evaluation is not required, provide an overall justification for that determination.
see attached

Continued ☒

VIII **Signoffs:** Preparer: [redacted] [redacted] Date: 2/28/07

50.59 SCREEN

(Print name) (Sign)

Qualified Reviewer: [redacted] [redacted] Date: 3 / 1 / 07

(Print name) (Sign)

IX. Has a copy of the completed screen been provided to Licensing? ☒ YES

SCREENING QUESTIONS

10CFR50.59 screening question 1 is answered "No" since the proposed procedure changes do not involve any physical alterations to the plant (no new or different type of equipment will be installed). There are no design changes involved to the reactor trip system (RTS), engineered safety features actuation system (ESFAS), or any other structures, systems, and components (SSCs).

10CFR50.59 screening question 2 is answered "No" since the proposed procedure changes do not adversely affect how any FSAR-described SSC design functions are performed or controlled. There are changes to the procedural controls for tripping the plant and isolating one of the FWIS functions, as discussed above. These changes deal with the performance of a manual reactor trip and the isolation of the FWIS that is derived from the coincidence of P-4 (satisfied by reactor trip breaker position switches showing the breakers are open after a reactor trip) and low reactor coolant system (RCS) T-avg (enabled at 564°F). The proper timing of when to manually trip the reactor has always been under licensee purview and requires no further evaluation under 50.59. The procedure changes related to blocking FWIS in MODE 2 have already been reviewed and approved by the NRC in OL Amendment 126.

10CFR50.59 screening question 3 is answered "No" since the proposed procedure changes do not revise or replace any FSAR-described accident evaluation methodology that is used to establish the design bases or used in the safety analyses. No changes to the Area 5 steamline break hazards analysis in FSAR Appendix 3B are required, as discussed further below in the response to screening question 5.

10CFR50.59 screening question 4 is answered "No" since the proposed procedure changes do not involve any tests or experiments not described in the FSAR.

10CFR50.59 screening question 5 is answered "No" since there are no changes required to the Technical Specifications (TS). TS Table 3.3.2-1 Function 8.a requires that the P-4 permissive be OPERABLE in MODES 1, 2, and 3. However, the operability of the P-4 permissive is unaffected by these procedure changes since nothing associated with the RTB position switches or SSPS cabinet design is being changed. The permissive itself will remain OPERABLE in MODES 1, 2, and 3; however, a non-critical enabled function "downstream" of P-4 will be blocked in MODE 2. This non-critical enabled function, FWIS on P-4 coincident with low RCS T-avg, is not a TS required SSC. If it were a TS required SSC, it would be required to be listed as a sub-function under TS Table 3.3.2-1 Function 5. It is not. FWIS on P-4 coincident with low RCS T-avg does not meet any of the four criteria for TS inclusion in 10 CFR 50.36(c)(2)(ii). This actuation signal is not required to mitigate any accident. This actuation signal is modeled in the analysis of an Area 5 steamline break, but only for the purposes of providing conservatively early feedwater isolation to minimize the time to SG tube uncovery for mass and energy (M/E) releases in Area 5 for break sizes ≤ 1.2 square feet per Section 6.6.2.2.3 and Table 6.6.2-4 of RSG WCAP-16140 and WCAP-16265. In other words, it is only modeled where it makes the results worse – it is not a required design function.

Therefore, a 50.59 Evaluation (form CA2512) is not required for the proposed procedure changes.

BACKGROUND

OTG-ZZ-00005 Addendum 1 Revision 0 was developed from steps in existing OTG-ZZ-00006 Addendum 3 and simulator scenarios as one of two methods, in the case of this addendum by performing a manual reactor trip, for inserting the control and shutdown banks. Some differences exist between this new OTG-ZZ-00005 Addendum 1 Revision 0 and existing OTG-ZZ-00006 Addendum 3 due to different entry conditions, including different operational MODES and the fact that the shutdown banks will now be inserted by the manual reactor trip (MODE 2 in the case of new OTG-ZZ-00005 Addendum 1 Revision 0 during a plant shutdown from 20% RTP to MODE 3 vs. a plant shutdown from MODE 3 to MODE 5 in the case of existing OTG-ZZ-00006 Addendum 3). However, these ancillary changes in the Purpose, Scope, Precautions and Limitations, and Prerequisites are to be expected. Changes to the Procedure Instructions are also required to reflect the different plant conditions that will exist at addendum entry (such as a heightened concern over excessive RCS temperature reductions and measures enacted for letdown control, different desired SG levels, different set of main control boards alarms to be expected, checking that intermediate range NIS channels indicate a lowering neutron flux, expectation that E-0 (Reactor Trip or Safety Injection), be entered if unexpected conditions arise, and additional steps tied to restoring and stabilizing SG level, RCS pressure, and RCS temperature). All of these changes are also to be expected given the different set of plant conditions associated with addendum usage. Changes have also been made to replace a specific SG narrow range level in some steps with a reference to the control or program band, but these are equivalent changes requiring no further screening.

Therefore, the timing of rod insertion and blocking an automatic feedwater isolation signal is at the crux of this 50.59 screening form.

In the NRC's Safety Evaluation for OL Amendment 126 dated 4-23-98, the NRC staff specifically reviewed and found acceptable our feedwater isolation signal (FWIS) bypass switch design and our using it to block the FWIS initiated by the coincidence of P-4 and low T-avg as long as its use was limited to the following plant conditions:

- T-avg less than or equal to 564°F (the plant can be in MODE 1 or 2, but T-avg must be ≤ 564°F)
- just prior to opening the reactor trip breakers (RTBs) which satisfies the P-4 portion of this feedwater isolation signal's logic.

These limitations will be met by step 5.2.11.b of OTG-ZZ-00005 Revision 25 which is being issued concurrently with this OTG-ZZ-00005 Addendum 1 Revision 0.

NRC also wanted this particular FWIS function to be restored by defeating the bypass prior to entering MODE 2 ascending during startup from an outage. This limitation is met by step 4.16 of existing OTG-ZZ-00002 Revision 36. As long as these limitations are observed, the plant will operate within the bounds of an amendment previously reviewed and approved by NRC.

This page and the following 4 pages are taken from NRC IR 05000482/2009004 which is found in ADAMS as ML093140803.

The cause of the finding has a problem identification and resolution crosscutting aspect in the area associated with the corrective action program because Wolf Creek failed to thoroughly evaluate the failure mechanism such that the resolutions address the causes and extent of conditions, as necessary. Specifically Wolf Creek did not properly consider the possibility of common-cause pitting failures which could have impacted the essential service water piping Train A structural integrity thereby affecting its cooling loads, including the Emergency Diesel Generator A [P.1(c)] (Section 1R15).

- Green. The inspectors identified a noncited violation of Technical Specification 3.8.1, Required Action B.4.2.2 on March 24, 2009 when the licensee performed elective maintenance on safety bus relays and removed equipment from service that was required by the technical specification and the NRC Safety Evaluation Report (SER) while in an extended diesel generator outage. The maintenance had the potential to open the normal offsite feeder breaker. This issue has been entered into the corrective action program as Condition Report 15727.

  The inspectors determined that the failure to implement requirements of Technical Specification 3.8.1 and the associated NRC safety evaluation was a performance deficiency. The finding was more than minor because it is associated with the equipment performance attribute for the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding was determined to be of very low safety significance because the issue did not result in the Train B offsite power being inoperable for greater than 24 hours and did not involve external events such as flooding. Additionally, the cause of the finding has a problem identification and resolution crosscutting aspect in the area associated with the corrective action program. Specifically, Wolf Creek did an extent of condition review in response to a previous violation which included Procedure STS IC-208B, but still failed to prohibit performance of STS IC-208B during the 7-day diesel outages [P.1(c)] (Section 1R19).

- Green. On August 22, 2009, the inspectors identified a noncited violation of Technical Specification 3.0.3 in which both trains of Technical Specification 3.3.2 engineered safety features actuation system interlock function 8.a were bypassed with jumper wires in accordance with a plant procedure. Function 8.a is the interlock for reactor trip signal coincident with lo Tave signal. Wolf Creek blocked the signal from the feedwater valves with jumper wires during control rod drive motor-generator testing in Mode 3. The inspectors and the NRR technical specification branch found this to be contrary to the Updated Safety Analysis Report, the technical specifications, the technical specification bases, and the NRC safety evaluations supporting the technical specifications. The licensee entered this issue in their corrective action program as Condition Report 19318.

  The inspectors found that the failure to implement Technical Specification 3.3.2 interlock, function 8.a was a performance deficiency. The inspectors determined that this finding was more than minor because it is associated with the design control attribute of the Mitigating Systems Cornerstone and it affected the cornerstone objective to ensure the availability, reliability, and capability of mitigating systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," and screened the finding to Phase 2 because the finding represents a loss of a system's function. The inspectors used Inspection Manual Chapter 0609, Appendix A and screened the finding to the NRC senior reactor analyst for review because there was not an acceptable equipment deficiency in the pre-solved worksheet. The senior reactor analyst determined that the finding is Green because he solved Table 3.10 of the Risk-Informed Inspection Notebook for Wolf Creek Generating Station, Revision 2.1a and found that the loss of feedwater isolation signal for less than 3 days resulted in a 1E-7 (Green) outcome. The inspectors also determined that the cause of the finding has a crosscutting aspect in the human performance area associated with decision making because Wolf Creek failed to make a risk significant decision using a systematic process. This issue was evaluated more than once and those evaluations sought to justify bypassing the interlock rather than seek the full regulatory basis for the interlock [H.1.a] (1R15).

- Green. The inspectors identified a noncited violation of 10 CFR 50 Appendix B, Criterion III, "Design Control," for failing to translate the boric acid design basis into procedures that ensure time sensitive operator actions are completed to achieve the core shutdown margin specified in the core operating limits report. Performance Improvement Request 2005-3461 identified that if the room coolers were started while lake temperature was low, the boric acid solution temperature may decrease below the solubility limit. Corrective actions for heat tracing and room temperature logging took approximately 3 years to implement and stopped short of addressing boric acid system operation when nonsafety power is lost to the heat tracing and the plant must be taken to cold shutdown in accordance with technical specifications. The licensee entered this issue in their corrective action program as Condition Report 20717.

The failure to translate the design bases into procedures that ensure the function of the safety-related boric acid system upon loss of nonsafety-related heat tracing is a performance deficiency. The inspectors determined that this finding was more than minor because this issue aligned with Inspection Manual Chapter 0612, Appendix E, example 2.f, because the pipe temperature was required to stay above the boric acid solubility limit and the loss of the heat tracing and or room temperature decrease will block the boric acid system. This issue was associated with the equipment performance attribute of the mitigating systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At Power Situations," and determined that the finding screened to phase 2 because the issue was a design or qualification deficiency confirmed to result in loss of operability or functionality The inspectors evaluated the significance of this finding using Phase 2 of Inspection Manual Chapter 0609, Risk Informed Inspection Notebook for Wolf Creek Generating Station, and determined that the finding was of very low safety significance because loss of the boric acid system in Table 3.9 for one year resulted in a 1E-7 CDF when giving recovery credit for the refueling water storage tank. The inspectors determined that this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Wolf Creek did not take appropriate corrective actions to resolve known deficiencies in the design and operation of the boric acid system for

.2 Introduction. On August 22, 2009, the inspectors identified a violation of Technical Specification 3.0.3 in which both trains of a Technical Specification 3.3.2 interlock in the engineered safety features actuation system were bypassed with jumper wires in accordance with plant procedure.

Description. On August 22, 2009, the inspectors observed that both trains of Technical Specification 3.3.2, function 8.a, P-4, were bypassed while in Mode 3. The inspectors found that Wolf Creek installed jumper wires on both trains in accordance with Procedure SYS SB-122, "Enabling/Disabling P-4/Lo Tave FWIS [feed water isolation signal]." The inspectors found that Wolf Creek has installed the jumper wires on both trains in the past to support reactor trip breaker and control rod drop testing in Mode 3. The jumpers defeated the function of both trains of reset switches on the main control board such that a P4/FWIS cannot be sent to close feedwater valves and trip the main feedwater pumps.

The inspectors reviewed the technical specification bases for the engineered safety features actuation system interlocks and function 8.a. The bases and USAR state that the functions of the interlock are to: 1) trip the main turbine, 2) isolate main feed water coincident with lo Tavg, 3) allow manual block of the automatic re-actuation of safety injection after a manual reset of safety injection, 4) allow arming of the steam dump valves and transfer the steam dump from the load rejection Tavg controller to the plant trip controller, 5) prevents opening of the main feed water isolation valves if they were closed on safety injection or steam generator hi-hi water level. The inspectors found that this was consistent with the standard improved technical specifications for Westinghouse plants and the Wolf Creek USAR, Table 7.3-15, "NSSS Interlocks for Engineered Safety Feature Actuation System." Under License Amendment 123, Wolf Creek converted to improved standard technical specifications in December 1999. The P-4 interlock description has not changed since 1999. The licensee submittals acknowledged that the functions of P-4 were not part of a design basis analysis, but were retained in the technical specifications to limit reactor coolant system cooldown following a reactor trip. Technical Specification 3.3.2 states that "The ESFAS [engineered safety features actuation signal] instrumentation for each Function in Table 3.3.2 shall be OPERABLE According to Table 3.3.2-1." Function 8 of Table 3.2.-1 covers interlocks and specifically interlock 8.a, P-4, is required to be Operable in Modes 1, 2, and 3. The inspectors found that function 8.a is required in Modes 1, 2, and 3. The inspectors consulted with the Office of Nuclear Reactor Regulation's technical specification branch and found that statements in the bases provide a summary of the technical specification and do not override requirements. The sentence in the bases that states: "This Function must be OPERABLE in MODES 1, 2, and 3 when the reactor may be critical or approaching criticality," clarifies why it is required in Modes 1, 2, and 3 and does not permit P-4 to be inoperable if the reactor is not approaching criticality. Operators are trained to anticipate criticality such as during control rod-drive motor-generator testing during August 22-23, 2009.

During interviews, Wolf Creek stated that it was necessary to bypass the P4/FWIS in order to perform rod-drive motor-generator set testing that cycled the reactor trip breakers. Wolf Creek contended that the P-4/FWIS was not necessary to assure compliance with the plant safety analysis. Lastly, Wolf Creek stated that during Mode 3 after refueling outages, it was necessary to install jumpers and bypass the P-4/FWIS for rod-drop testing because operation of the main feedwater system in automatic level control was more desirable than having an operator manually control steam generator levels with auxiliary feedwater. The inspectors agreed that this interlock is not assumed in Chapter 15 of the USAR, but the inspectors found that the Wolf Creek technical specification bases state that "ESFAS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii)" which is identical to the generic standard specifications approved by the NRC. The inspectors found that there are several technical specification systems such as steam generator atmospheric relief valves, the condensate storage tank, and pressurizer power operated relief valves that are not in Chapter 15 of the USAR but are required to be operable under technical specifications per 10 CFR 50.36. Thus, the inspectors found that the interlock's absence in Chapter 15 of the USAR does not mean it is not required by the technical specification. Wolf Creek previously evaluated this condition in Performance Improvement Request 2001-0041 which concluded this P-4/FWIS was not required to be operable in any Mode because it is not credited in Chapter 15 of the USAR. Wolf Creek also used other plants with NRC approved safety evaluations to justify the use of Procedure SYS SB-122 rather than requesting a license amendment. The inspectors found that these conclusions are incorrect.

The inspectors found that control room operators did not log the inoperability of P-4 until after inspector questioning, and afterward, operators incorrectly applied Technical Specification 3.3.2, Condition F, which allowed 60 hours to return one train of the interlock to service. With both trains of P4 bypassed, Technical Specification 3.0.3 applied and Wolf Creek had 13 hours to be in Mode 4. The P-4 interlock was inoperable for approximately 20 hours from August 22-23, 2009. Wolf Creek missed the transition to Mode 4.

Analysis. The inspectors found that the failure to evaluate implement Technical Specification 3.3.2 interlock, function 8.a was a performance deficiency. The inspectors determined that this finding was more than minor because it is associated with the design control attribute of the Mitigating Systems Cornerstone and it affected the cornerstone objective to ensure the availability, reliability, and capability of mitigating systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609.04, 'Phase 1 - Initial Screening and Characterization of Findings," and screened the finding to Phase 2 because the finding represents a loss of a system's function. The inspectors used Inspection Manual Chapter 0609, Appendix A and screened the finding to the NRC senior reactor analyst for review because there was not an acceptable equipment deficiency in the pre-solved worksheet. The senior reactor analyst determined that the finding is Green because he solved Table 3.10 of the Risk-Informed Inspection Notebook for Wolf Creek Generating Station, Revision 2.1a and found that the loss of feedwater isolation signal for less than 3 days resulted in a 1E-7 (Green) outcome. The inspectors also determined that the cause of the finding has a crosscutting aspect in the human performance area associated with decision making because Wolf Creek failed to make a risk significant decision using a systematic process. This issue was evaluated more than once and those evaluations sought to justify bypassing the interlock rather than seek the full regulatory basis for the interlock. [H.1.a]

Enforcement. Wolf Creek Technical Specification, Table 3.3.2.1, function 8 includes engineered safety features actuation system interlocks. Function 8.a, the P-4 interlock, requires two trains to be operable in Modes 1, 2, and 3. Function 8.a does not provide a required action for both trains of engineered safety features actuation system interlocks inoperable. Wolf Creek Technical Specification 3.0.3 requires the plant to be in Mode 4 within 13 hours if there is no required action specified for a limiting condition of operation that cannot be met. Contrary to the above, from August 22 to August 23, 2009, Wolf Creek failed to change modes from Mode 3 to Mode 4 when both trains of engineered safety features actuation system interlock function 8.a, P-4, were inoperable for greater than 13 hours. Specifically, from August 22 to 23, 2009, Wolf Creek failed to change modes from Mode 3 to Mode 4 when both trains were removed from service for approximately 20 hours. Because this violation was determined to be of very low safety significance and was placed in the corrective action program as Condition Report 19318, this violation is being treated as a noncited violation in accordance with Section VI.A.1 of the Enforcement Policy: NCV 05000482/2009004-04, "Failure to Implement Engineered Safety Features Actuation System Technical Specification Results in Missed Mode Change."

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed the effectiveness of the licensee's implementation of evaluations performed in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments," and changes, tests, experiments, or methodology changes that the licensee determined did not require 10 CFR 50.59 evaluations. The inspection procedure requires the review of 6 to 12 licensee evaluations required by 10 CFR 50.59, 12 to 25 changes, tests, or experiments that were screened out by the licensee and 5 to 15 permanent plant modifications.

The inspectors reviewed 9 evaluations required by 10 CFR 50.59. These included:

- 2006-001, Radiological Consequences of a Fuel Handling Accident, Revision 0
- 2008-0006, Wolf Creek Generating Station (WCGS) Simplified Head Assembly (SHA) Drop Analysis, Revision 0
- 2008-0008, Use of Dedicated Operator for SI Pump B Room cooler Replacement, Revision 0
- 2005-004, WCGS Rod Withdrawal at Power Event Safety Analysis, Revision 0
- 2008-001, Evaluations of Voids in the ECCS Suction Piping, Revision 0
- 2008-002, Evaluations of Voids in the ECCS Discharge Piping, Revision 0
- 2006-002, Power Operation, Revision 54

This page is taken from NRC Integrated Inspection Report 05000482/2009005 which is found in ADAMS as ML100430713.

The finding was more than minor because it was associated with the configuration control (reactivity control) attribute of the Barrier Integrity Cornerstone, and it affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter 0609.04, and determined that the finding screened to Green because the P-6 interlock only affected the fuel barrier (Section 4OA2). This finding was not assigned a crosscutting aspect because the cause was not representative of current performance.

Cornerstone: Occupational Radiation Safety

- Green. The inspector identified a noncited violation of Technical Specification 5.7.2.a.1 for failure to maintain administrative control of door and gate keys to high radiation areas with dose rates greater than 1 rem per hour but less than 500 rads per hour (referred to as locked high radiation areas). Specifically, as of October 21, 2009, the licensee did not have administrative controls over a single master key to locked high radiation areas. This issue was entered into the licensee's corrective action program as Condition Report 20973.

  Failure to maintain administrative control of the master key to locked high radiation areas was a performance deficiency. This finding is greater than minor because if left uncorrected the finding has the potential to lead to a more significant safety concern in that an individual could receive unanticipated radiation dose by gaining access a locked high radiation area without the proper controls and briefing. This finding was evaluated using the occupational radiation safety significance determination process and determined to be of very low safety significance because it did not involve: (1) as low as is reasonably achievable planning or work control issue, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. Additionally, the violation has a crosscutting aspect in the area of human performance associated with the work practices component because the lack of peer and self-checking resulted in inadequate control of keys to locked high radiation areas [H.4(a)] (Section 2OS1).

Cornerstone: Miscellaneous

- Severity Level IV. The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.73 in which the licensee failed to submit a licensee event report within 60 days following discovery of events or conditions meeting the reportability criteria. On December 31, 2009, the inspectors identified a licensee event report that was no timely. Licensee Event Report 2009-009-00 was not issued within 60 days for a condition prohibited by technical specifications, and the event report did not identify that the disabling of both trains of the P-4 interlock on August 22, 2009 was also reportable per 10 CFR 50.73(a)(2)(v). The P-4 interlock was required by Technical Specification 3.3.2, function 8.a, and is discussed in USAR, Section 7.3.8, "NSSS Engineered Safety Feature Actuation System." Wolf Creek licensee event report 2009-009 was correct in that the interlock is not credited in accident analysis. However, NUREG 1022, Section 3.2.6, specifies that inoperable systems required by the technical specifications be reported, even if there are other diverse operable means of accomplishing the safety function.

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION

This LER is in ADAMS as ML100890421.

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
WOLF CREEK GENERATING STATION	05000 482	1 OF 5

4. TITLE
Defeating Feedwater Isolation on Low Tavg Coincident with P-4 Function Results in Missed Mode Change

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
08	22	2009	2009	009	01	03	22	2010	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 3

10. POWER LEVEL: 000

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: (Check all that apply)

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☒ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☒ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Richard D. Flannigan, Manager Regulatory Affairs	(620) 364-4117

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (If yes, complete 15. EXPECTED SUBMISSION DATE)	☒ NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On August 22, 2009, at 0540 hours Central Daylight Time (CDT), with the plant in Mode 3, Control Room staff defeated the feedwater isolation on low Tavg coincident with P-4 Function using procedure SYS SB-122, "Enabling/Disabling P-4/Lo Tavg FWIS." This procedure was performed for restoring main feedwater flow through the main feedwater isolation valves (MFIVs) to supply water to the steam generators. On August 23, 2009, at 0125 hours the jumpers installed for defeating the feedwater isolation on low Tavg coincident with P-4 function were removed and procedure SYS SB-122 completed at 0140 hours.

The Nuclear Regulatory Commission (NRC) Resident questioned the defeating of the feedwater isolation on low Tavg coincident with P-4 function while in Mode 3. Technical Specification (TS) 3.3.2, Table 3.3.2-1 specifies the applicable Mode for Function 8.a. (Reactor Trip, P-4) as Modes 1, 2, 3. Defeating the feedwater isolation on low Tavg coincident with P-4 function using procedure SYS SB-122 results in both channels being defeated. There is no TS Condition for two inoperable trains. Limiting Condition for Operation (LCO) 3.0.3 specifies that when an associated Action is not provided, action shall be initiated within 1 hour to place the plant in Mode 4 in 13 hours. Action was not taken as required by the TSs.

WOLF CREEK
NUCLEAR OPERATING CORPORATION

This document is found in ADAMS as ML101100391. This is the document whereby Wolf Creek formally requested to change its Technical Specification to allow blocking the Lo-Tavg/P4 FWIS during MODE 3.

Terry J. Garrett
Vice President Engineering

April 13, 2010

ET 10-0014

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Docket No. 50-482: Application To Revise Technical Specification 3.3.2, "Engineered Safety Feature Actuation System Instrumentation," Table 3.3.2-1

Gentlemen:

Pursuant to 10 CFR 50.90, Wolf Creek Nuclear Operating Corporation (WCNOC) hereby requests an amendment to Renewed Facility Operating License No. NPF-42 for the Wolf Creek Generating Station (WCGS). The proposed amendment revises Table 3.3.2-1, Function 8.a., (ESFAS Interlocks, Reactor Trip, P-4) of Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System Instrumentation." WCNOC is proposing to add footnote (m) to Function 8.a. to identify the enabled functions and the applicable MODES for the Reactor Trip, P-4 Interlock Function.

Attachment I through IV provide the Evaluation, Markup of TSs, Retyped TS pages, and proposed TS Bases changes, respectively, in support of this amendment request. Attachment IV, proposed changes to the TS Bases, is provided for information only. Final TS Bases changes will be implemented pursuant to TS 5.5.14, "Technical Specification (TS) Bases Control Program," at the time the amendment is implemented. Attachment V provides a List of Regulatory Commitments made by WCNOC in this submittal.

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

This amendment application was reviewed by the Plant Safety Review Committee. In accordance with 10 CFR 50.91, a copy of this amendment application, with attachments, is being provided to the designated Kansas State official.

A001
NRR

P.O. Box 411 / Burlington, KS 66839 / Phone: (620) 364-8831

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 30, 2011

This document is found in ADAMS as ML110550846. This is the document whereby the US NRC granted Wolf Creek permission to change their Technical Specifications such that the Lo-Tavg/P-4 FWIS could be blocked in MODE 3.

Mr. Matthew W. Sunseri
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: REVISE TABLE 3.3.2-1 OF TECHNICAL SPECIFICATION 3.3.2, "ENGINEERED SAFETY FEATURE ACTUATION SYSTEM (ESFAS) INSTRUMENTATION" (TAC NO. ME3762)

Dear Mr. Sunseri:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 194 to Renewed Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 13, 2010, as supplemented by letters dated October 13 and December 21, 2010, and January 18, 2011.

The amendment revises TS Table 3.3.2-1, Function 8.a (Reactor Trip, P-4) by adding footnote (m) to identify the enabled functions and the applicable modes for the Reactor Trip, P-4 interlock function.

A copy of our related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

Balwant K. Singal

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures:
1. Amendment No. 194 to NPF-42
2. Safety Evaluation

cc w/encls: Distribution via Listserv

Table 3.3.2-1 (page 5 of 5)
Engineered Safety Feature Actuation System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
7.	Automatic Switchover to Containment Sump					
	a. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
	b. Refueling Water Storage Tank (RWST) Level - Low Low	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 35.5% of instrument span
	Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
8.	ESFAS Interlocks					
	a. Reactor Trip, P-4[m]	1,2,3	2 per train, 2 trains	F	SR 3.3.2.11	NA
	b. Pressurizer Pressure, P-11	1,2,3	3	L	SR 3.3.2.5 SR 3.3.2.9	≤ 1979 psig

(a) The Allowable Value defines the Limiting Safety System Settings. See the Bases for the Trip Setpoints.

(m) The functions of the Reactor Trip, P-4 interlock required to meet the LCO are:

- Trips the main turbine – MODES 1 and 2
- Isolates MFW with coincident low T_{avg} – MODES 1 and 2
- Allows manual block of the automatic reactuation of SI after a manual reset of SI – MODES 1, 2, and 3
- Prevents opening of MFIVs if closed on SI or SG Water Level – High High – MODES 1, 2, and 3

This is page 6 of ML110550846. It shows how the Wolf Creek Technical Specifications were revised under Amendment 196 to allow Wolf Creek to disable the P-4/564°F FWIS during MODE 3.

This sheet can be found in ADAMS as page 63 of ML111661877. It shows how Table 3.3.2-1 function 8.a. appears in Callaway Plant's Technical Specifications as of July 29, 2011.

Table 3.3.2-1 (page 10 of 11)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.13	NA
b. Refueling Water Storage Tank (RWST) Level - Low Low	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 35.2%
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
8. ESFAS Interlocks					
a. Reactor Trip, P-4	1,2,3	2 per train, 2 trains	F	SR 3.3.2.11	NA
b. Pressurizer Pressure, P-11	1,2,3	3	L	SR 3.3.2.5 SR 3.3.2.9	≤ 1981 psig
9. Automatic Pressurizer PORV Actuation					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.14	NA
b. Pressurizer Pressure – High	1,2,3	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9	≤2350 psig

(a) The Allowable Value defines the limiting safety system setting except for Functions 1.e, 4.e.(1), 5.c, 5.e.(1), 5.e.(2), 6.d.(1), and 6.d.(2) (the Nominal Trip Setpoint defines the limiting safety system setting for these Functions). See the Bases for the Nominal Trip Setpoints.

Note that, just like at Wolf Creek prior to Amendment No. 194, the Callaway Plant Technical Specification for function 8.a is applicable in MODE 3.